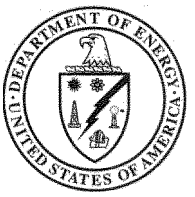
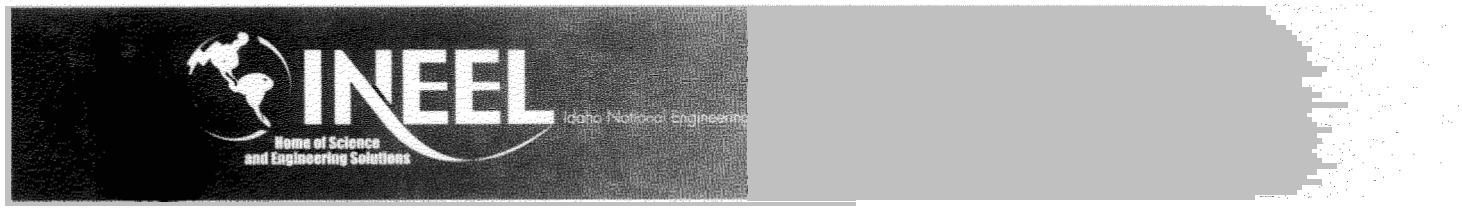
DOE/NE-ID-11187
Revision A
August 2004



U.S. Department of Energy
Idaho Operations Office

Engineering Evaluation/Cost Analysis for the Power Burst Facility Reactor Building Non-Time Critical Removal Action, Idaho National Engineering and Environmental Laboratory (Draft)



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Revision A
Project No. 24128

Engineering Evaluation/Cost Analysis for the Power Burst Facility Reactor Building Non-Time Critical Removal Action, Idaho National Engineering and Environmental Laboratory (Draft)

August 2004

Prepared for the
U.S. Department of Energy
Idaho Operations Office

ABSTRACT

This Engineering Evaluation/Cost Analysis assists the U.S. Department of Energy Idaho Operations Office in identifying the preferred response alternative for the Power Burst Facility reactor building (PER-620). It is intended to (1) satisfy environmental review requirements for the removal action, (2) provide a framework for evaluating and selecting alternative technologies, and (3) satisfy Administrative Record requirements for documentation of the removal action selection. This Engineering Evaluation/Cost Analysis identifies the objectives of the removal action and analyzes the effectiveness, implementability, and cost of various alternatives that could satisfy these objectives.

EXECUTIVE SUMMARY

This Engineering Evaluation/Cost Analysis has been prepared for public comment. It considers five alternatives for the decontamination and decommissioning of the Power Burst Facility (PBF) at the Idaho National Engineering and Environmental Laboratory (INEEL). The PBF reactor operated from 1972 to 1985 to conduct tests of reactor fuel in extreme environments. The nuclear fuel was removed in 2003 and actions related to potential hazardous waste covered in the Voluntary Consent Order (including removal of 38,000 lb of shielding lead) have been completed. The U.S. Department of Energy Idaho Operations Office (DOE-ID) is now ready to decommission the facility. The scope of this removal action includes actions on the PBF reactor building (PER-620).

The five alternatives range from no action (Alternative 5) to the removal of all radionuclide and nonradionuclide inventory from the facility (Alternative 3). In between are alternatives that involve grouting the belowground structure in place (Alternative 1) or removing some of the shielding lead inventory prior to grouting the belowground structure in place (Alternative 2). Alternative 4 comprises temporary stabilization of the facility with long-term surveillance and monitoring.

One of the key issues in this removal action is the presence of 13 m^3 of lead in the facility substructure. Much of it would be difficult to remove from the structure and would result in considerable worker risk, including radiation exposure, if removed. Another key issue is the amount of worker risk and radiation exposure that would be associated with the removal of the reactor and associated activated materials. The risk assessment illuminates the tradeoff in short-term versus long-term risks between either removing the lead and activated materials from the substructure in order to reduce the remaining radionuclide and nonradionuclide inventory or leaving them in place in order to avoid worker risk, including radiation exposure. The risk assessment described in this document demonstrates that leaving all the contaminants in place in the building substructure, after removal of the aboveground structure and all water, would not pose unacceptable risk to groundwater nor would it cause the Idaho Ground Water Quality standards (maximum contaminant levels) to be exceeded.

The report recommends Alternative 1—removal and disposal of water in tanks and piping, followed by grouting the remaining substructure and contents in place, removal and disposal of the aboveground structure, and installation of a Resource Conservation and Recovery Act performance-based cover. Alternative 1 also includes meeting the landfill postclosure requirements—either through incorporation in the design of this removal action as applicable or relevant and appropriate requirements under the Comprehensive Environmental Response, Compensation, and Liability Act (42 USC § 9601 et seq.) or through submittal of a Hazardous Waste Management Act/Resource Conservation and Recovery Act (Idaho Code 39-4401 et seq.; 42 USC § 6901 et seq.) postclosure permit application—and a commitment to long-term monitoring in either case.

The facility water would be primarily disposed of in the evaporation ponds at the INEEL CERCLA Disposal Facility or at the Test Reactor Area evaporation ponds. The debris and contents of the abovegrade structure would be disposed of at the INEEL CERCLA Disposal Facility, Radioactive Waste Management Complex, or Central Facilities Area Industrial Landfill Complex, depending on the waste characteristics.

The recommended alternative meets the proposed removal action objectives regarding long-term risk, minimizes short-term worker risk and radiation exposure (1.2 person-rem), is cost-effective ($8.4 million in net present value costs), and provides a safe, stable, and permanent configuration that is environmentally sound. The alternative can be implemented in less than 1 year and permits the DOE-ID to complete closure actions at the PBF area site, thereby permitting focus on other cleanup, closure, and new mission activities.

Although the PBF reactor was not specifically addressed in the *Record of Decision Power Burst Facility and Auxiliary Reactor Area, Operable Unit 5-12,* the action is consistent with the remedial action objectives for soil sites and the future land use assumptions in the Record of Decision. It is also consistent with past actions taken on reactor facilities in the PBF area.

The DOE-ID also considers Alternative 2 reasonable within the range of acceptable alternatives. Alternative 2 would accomplish removal of approximately 2/3 of the lead inventory. However, because there would be no benefit achieved in reduced risk to groundwater or direct exposure risk to compensate for the substantially higher estimated worker radiation exposure (9.3 person-rem) and cost ($4.0 million additional net present value costs), this alternative is inconsistent with the as-low-as reasonably achievable (ALARA) principles for radiation workers. For these reasons, the DOE-ID prefers Alternative 1.

The DOE-ID considers Alternatives 3, 4, and 5 unacceptable. Alternative 3 would result in a worker exposure of up to 155 person-rem with major technology development needed to reduce those risk levels. The removal of all lead and radionuclide inventory is not practical because of the high worker exposures, technology development needs, and cost ($17 million in net present value costs). Furthermore, there would be no groundwater or direct exposure risk reduction benefit achieved for the substantially higher worker risks and costs. Alternative 4 is unacceptable, because it would simply stabilize the facility, delaying final closure to the future. Alternative 5 (no action) is not acceptable, because it would not meet the removal action objectives.

This action is being proposed under a non-time critical removal action. Under a non-time critical removal action, a removal action can be taken to abate, prevent, minimize, stabilize, mitigate, or reduce the release or threat of release of contaminants. An engineering evaluation/cost analysis is required under 40 *Code of Federal Regulations* 300.415(b)(4)(i) of the "National Oil and Hazardous Substances Pollution Contingency Plan" for all non-time critical removal actions.

This Engineering Evaluation/Cost Analysis will become part of the INEEL Administrative Record. It will be made available for public comment. The INEEL Administrative Record is on the Internet at http://ar.inel.gov/ and is available to the public at the following locations:

Albertson's Library
Boise State University
1910 University Drive
Boise, ID 83725
(208) 426-1625

INEEL Technical Library
DOE Public Reading Room
1776 Science Center Drive
Idaho Falls, ID 83415
(208) 526-1185

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ACRONYMS

ALARA	as low as reasonably achievable
ARA	Auxiliary Reactor Area
ARAR	applicable or relevant and appropriate requirement
CERCLA	Comprehensive Environmental Response, Compensation, and Liability Act
CFA	Central Facilities Area
CFR	Code of Federal Regulations
D&D	decontamination and decommissioning
DCF	dose conversion factor
DEQ	Idaho Department of Environmental Quality
DOE	U.S. Department of Energy
DOE-ID	U.S. Department of Energy Idaho Operations Office
EDF	engineering design file
EE/CA	engineering evaluation/cost analysis
EPA	U.S. Environmental Protection Agency
FY	fiscal year
HVAC	heating, ventilation, and air conditioning
HWMA	Hazardous Waste Management Act
IC	institutional control
ICDF	INEEL CERCLA Disposal Facility
ICP	Idaho Completion Project
IDAPA	Idaho Administrative Procedures Act
INEEL	Idaho National Engineering and Environmental Laboratory
IPT	inpile tube
ISMS	Integrated Safety Management System
K_d	sorption coefficient

MCL	maximum contaminant level
NA	not applicable
NCRP	National Council on Radiation Protection
O&M	operations and maintenance
OU	operable unit
PBF	Power Burst Facility
PCB	polychlorinated biphenyl
PPE	personal protective equipment
RCRA	Resource Conservation and Recovery Act
RI/FS	remedial investigation/feasibility study
RWMC	Radioactive Waste Management Complex
SPERT	Special Power Excursion Reactor Test
SSSTF	Staging, Storage, Sizing, and Treatment Facility
TBC	to be considered
TRA	Test Reactor Area
USC	United States Code
VCO	Voluntary Consent Order
WAG	waste area group
WERF	Waste Experimental Reduction Facility
WROC	Waste Reduction Operations Complex

Engineering Evaluation/Cost Analysis for the Power Burst Facility Reactor Building Non-Time Critical Removal Action, Idaho National Engineering and Environmental Laboratory (Draft)

1. INTRODUCTION

This Engineering Evaluation/Cost Analysis (EE/CA)—prepared in accordance with Section 300.415(b)(4)(i) of the "National Oil and Hazardous Substances Pollution Contingency Plan" (40 CFR 300) assists the U.S. Department of Energy Idaho Operations Office (DOE-ID) in identifying the preferred response alternative for the Power Burst Facility (PBF) reactor building (PER-620). It is intended to (1) satisfy environmental review requirements for the removal action, (2) provide a framework for evaluating and selecting alternative technologies, and (3) satisfy Administrative Record requirements for documentation of the removal action selection. This EE/CA identifies the objectives of the removal action and analyzes the effectiveness, implementability, and cost of various alternatives that could satisfy these objectives.

Reactor fuel was removed from the facility in 2003. Efforts have been completed to remove other nonnuclear and nuclear facilities and structures in the PBF Complex area. Other activities have also been completed—and others are underway and planned—to remove remaining nonradioactive and radioactive water, materials, and debris from the PBF Complex area and PER-620 and in advance of decontamination and decommissioning (D&D) of the PER-620 PBF reactor building. Efforts have been completed to characterize the contents of the facility. The DOE-ID has chosen to move forward with D&D of the PBF reactor building through the Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA) non-time critical removal action process.

The PBF reactor building's non-time critical removal action comprises D&D of the PBF reactor building (PER-620). Currently, the canals, vessels, piping, and overall facility contain water that provides shielding for the reactor as well as residual fission product material, activated metals, and radioactive surface contamination. In addition, the facility contains elemental lead that was and is used for shielding. With issuance of an eventual action memorandum, the CERCLA removal action would commence and DOE-ID would declare the canals, vessels, piping, and reactor annulus no longer operable when the water is no longer needed for shielding. The removal action would place the facility in a configuration that is protective of human health and the environment. This action is consistent with the joint U.S. Department of Energy (DOE) and U.S. Environmental Protection Agency (EPA) *Policy on Decommissioning of Department of Energy Facilities Under the Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA)* (DOE and EPA 1995), which establishes the CERCLA non-time critical removal action process as the approach for decommissioning.

2. SITE CHARACTERIZATION

This section provides summary background and description for the PER-620 reactor building, identifies previous and ongoing closure and cleanup activities, and provides a summary of the completed radiological and nonradiological characterization of the building. Finally, this section provides a summary of the completed evaluation of groundwater pathway risks associated with those characterizations and inventories.

2.1 Site Description and Background

2.1.1 Idaho National Engineering and Environmental Laboratory

The Idaho National Engineering and Environmental Laboratory (INEEL), managed by DOE-ID, is a government facility located 51 km (32 mi) west of Idaho Falls, Idaho. The INEEL occupies 2,305 km^2 (890 mi^2) of the northeastern portion of the Eastern Snake River Plain. In 1949, the U.S. Atomic Energy Commission established the INEEL, which was called the National Reactor Testing Station at that time. Its purpose was to conduct nuclear energy research and related activities. It was re-designated the Idaho National Engineering Laboratory in 1974 and then the INEEL in 1997 to reflect expansion of its mission to include a broader range of engineering and environmental management activities.

The DOE-ID controls all land within the INEEL, and public access is restricted to public highways, DOE-ID-sponsored tours, special-use permits, and the Experimental Breeder Reactor I National Historic Landmark. In addition, DOE-ID accommodates Shoshone-Bannock tribal member access to areas on the INEEL for cultural and religious purposes.

The INEEL is located primarily in Butte County; however, it also occupies portions of Bingham, Bonneville, Clark, and Jefferson counties. The 2000 census indicated the following populations (in parentheses) for cities in the region: Idaho Falls (50,730), Pocatello (51,466), Blackfoot (10,419), Arco (1,026), and Atomic City (25).

Surface water flows on the INEEL consist mainly of three streams draining intermountain valleys to the north and northwest of the INEEL Site: (1) the Big Lost River, (2) the Little Lost River, and (3) Birch Creek. Flows from Birch Creek and the Little Lost River seldom reach the INEEL because of irrigation withdrawals upstream. The Big Lost River and Birch Creek usually flow onto the INEEL before the irrigation season or during high water years.

The physical characteristics, climate, flora and fauna, demography, and cultural resources of the INEEL are described in the *Record of Decision Power Burst Facility and Auxiliary Reactor Area, Operable Unit 5-12* (DOE-ID 2000).

2.1.2 Power Burst Facility Area

Once known as the Special Power Excursion Reactor Test (SPERT) facilities, the PBF area (Figure 1) consists of five separate operational areas: (1) the PBF control area, (2) the PBF reactor area (SPERT-I), (3) the Waste Engineering Development Facility (SPERT-II), (4) the Waste Experimental Reduction Facility (WERF) (SPERT-III), and (5) the Mixed Waste Storage Facility (SPERT-IV). Collectively, the WERF, Waste Engineering Development Facility, and the Mixed Waste Storage Facility were known as the Waste Reduction Operations Complex (WROC). It is located in the south-central region of the INEEL, approximately 9 mi east of the Central Facilities Area (CFA). At the PBF reactor area, the SPERT-I reactor was operated from 1955 to 1964. It was decommissioned in 1964 and demolished in 1985. The PBF reactor was constructed in 1972 just north of the remains of the old SPERT-I facility.

2.1.3 Power Burst Facility Reactor Facility

The PBF reactor operated from 1972 to 1985. Other structures in the vicinity include a maintenance and storage building, two electrical substations, and numerous smaller buildings and structures (Figure 2). Much of the information in this section has been extracted from Engineering

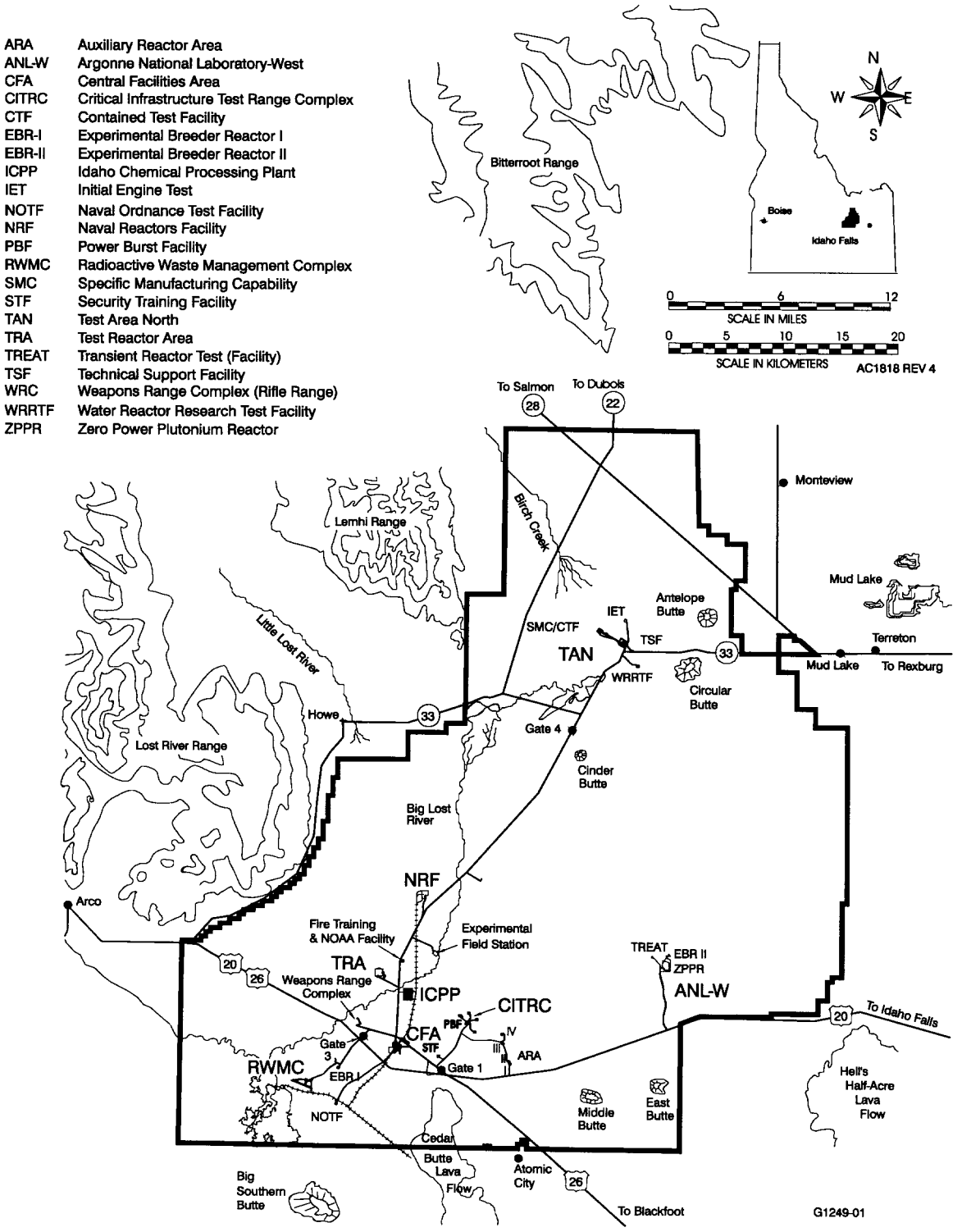

Figure 1. Location of the Power Burst Facility reactor area on the Idaho National Engineering and Environmental Laboratory Site.

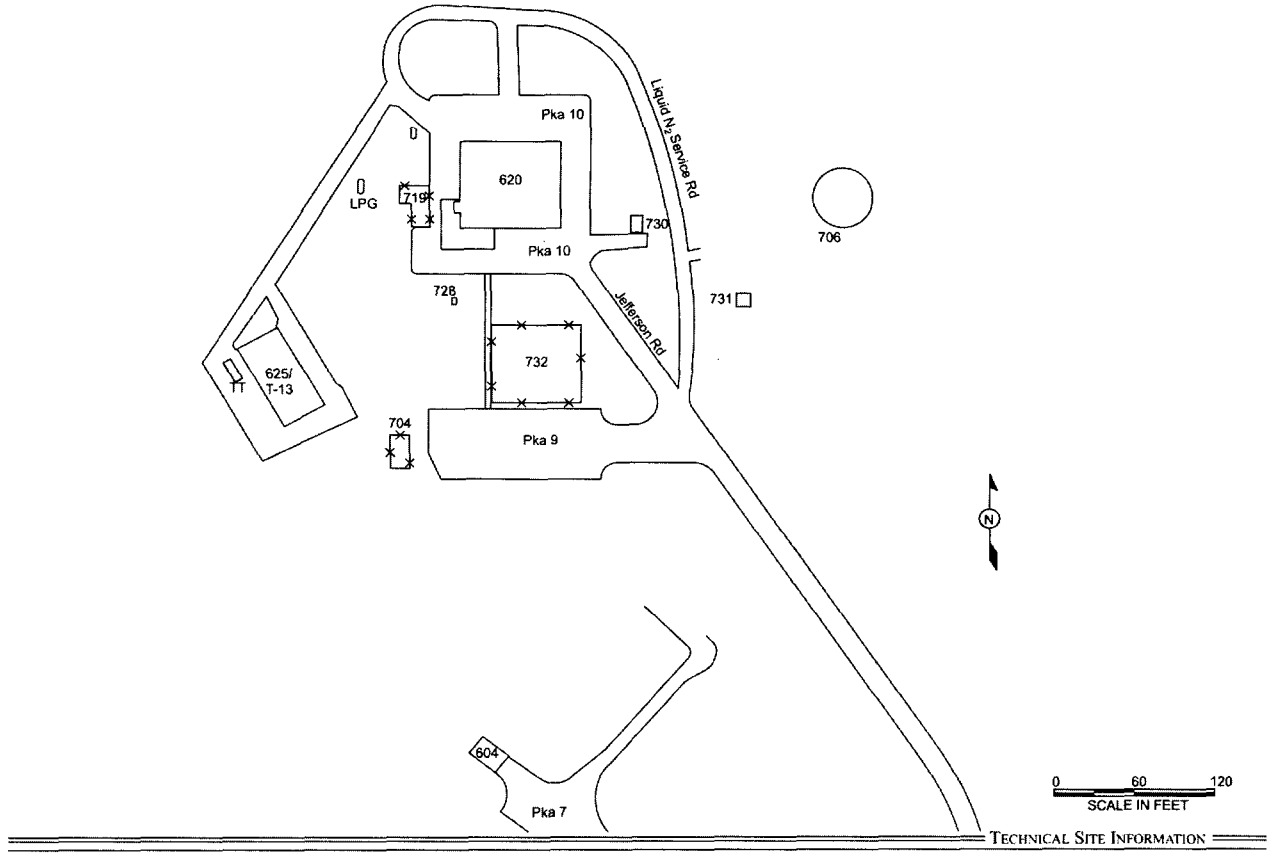

Figure 2. General layout of the Power Burst Facility part of the Experimental Reduction Facility/Waste Reduction Operations Complex/Power Burst Facility area.

Design File (EDF) –4697, "Radiological Characterization of the PBF Reactor for Disposal," and EDF-4943, "Nonradiological Inventory of Materials and Components in Subgrade Basement Levels/Areas of the Power Burst Facility Reactor Building (PER-620)."

Figure 2 shows the general layout of the PBF part of the WERF/WROC/PBF area. This EE/CA addresses the PBF reactor building (PER-620) at the PBF reactor area (SPERT-I) only (Figure 2); PER-620 is the PBF reactor building. It is shown in plan and elevation views in Figures 3 and 4.

The PBF reactor building houses the reactor vessel, fuel storage canal, and various process systems that supported reactor operations. The structure is a two-story steel-framed building with steel plate interior with aluminum exterior siding and two block-wall wings (east and west). The building is divided into a main reactor high-bay room, two single-story wings containing instrumentation and electrical control equipment, various support offices, operational and utility areas, and a two-level basement.

The main floor of the building contains the high bay; offices for the shift supervisor, operator training, and radiological control technicians; a decontamination room; a counting room; personal protective equipment (PPE) issue room; a tool crib; bathrooms; and change rooms. The high bay contains the canal (which joins the reactor on the south side), a 1-ton jib crane, and a 15-ton bridge crane. The high bay floor has hatches leading to Loop Cubicles 10 and 13. These cubicles contain nearly all the shielding lead in the building. Additional support and operational areas include the process control room and the

furnace and equipment room. The east wing of the main floor contains the mechanical work area, test loop control room, the experimental instrumentation room, and an electronic work area.

The building has two basement levels, which are connected by a stairwell and floor hatches. The first basement level contains part of the reactor vessel enclosure, Loop Cubicles 10 and 13, process and utility equipment, the experimental loop pipe access tunnel, and a sampling area. The second basement level contains the loop knockout drum room, subpile room, warm waste and hot waste room, poison injection system room, additional process and utility equipment, and the waste gas exhaust room.

Figure 3 depicts the subfloor chambers shown at the left or on the north side of the basement. The loop cubicle represents three chambers, one behind another. In this view, Cubicle 10 is closest to the viewer. The main function of this chamber was processing the experimental loop coolant. The sampling room is behind it, and, easternmost, Cubicle 13 is behind the sampling room, which housed the blowdown tank among other functions. The canal contains water used for shielding. Figure 4 shows Cubicles 10 and 13 in plan view.

Figure 5 shows an artist's rendering of the PBF reactor in sectional view. The reactor core is located centrally in a stainless-steel reactor vessel, which was filled with water. Experiments were contained in an Inconel 718 inpile tube (IPT) that occupied the central flux trap of the core and extended well above and below the core. The experimental test trains, after their use in the PBF core, were first held in the PBF canal and subsequently moved to the canal of the Materials Test Reactor and then to the Radioactive Waste Management Complex (RWMC).

The testing environment for the IPT was provided by the pressurized water coolant loop. The reactor core had an overall diameter of 1.32 m (52 in.), and it was 91 cm (36 in.) high. It contained 2,392 fuel rods and 104 shim rods. The fuel was enriched UO_2 (~18.5% U-235) diluted with calcium oxide-stabilized zircium oxide and clad with Type 304L stainless steel. Fuel rods were surrounded by a row of solid stainless-steel reflector rods and water. There were eight B4C control rods and four transient rods of similar construction used to control criticality and flux transients. The PBF fuel rods were removed in the summer of 2003.

2.2 Previous Closure/Cleanup Activities at the Power Burst Facility

The PBF reactor was placed on operational standby in 1985. The PBF fuel rods were removed in the summer of 2003.

The *Record of Decision Power Burst Facility and Auxiliary Reactor Area, Operable Unit 5-12* (DOE-ID 2000) selected a remedy for the cleanup of identified contaminated soil at PBF and the Auxiliary Reactor Area (ARA). Remedies also were selected for a radionuclide tank and a sanitary waste system at ARA. All remedial actions have been completed at PBF/ARA and, as required under CERCLA (42 USC § 9601 et seq.) whenever waste is left in place, institutional controls have been implemented for residual contaminants left in place at concentrations that would not allow for unrestricted use or access. Figure 6 shows the locations of current and planned institutionally controlled areas at PBF/ARA.

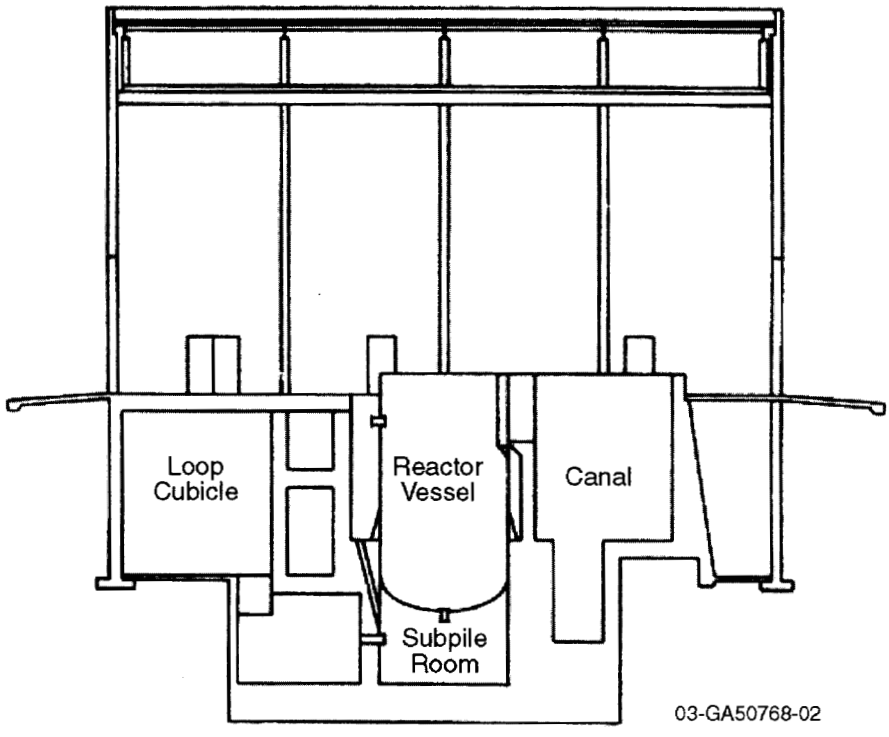

Figure 3. Power Burst Facility reactor building (PER-620) elevation looking east.

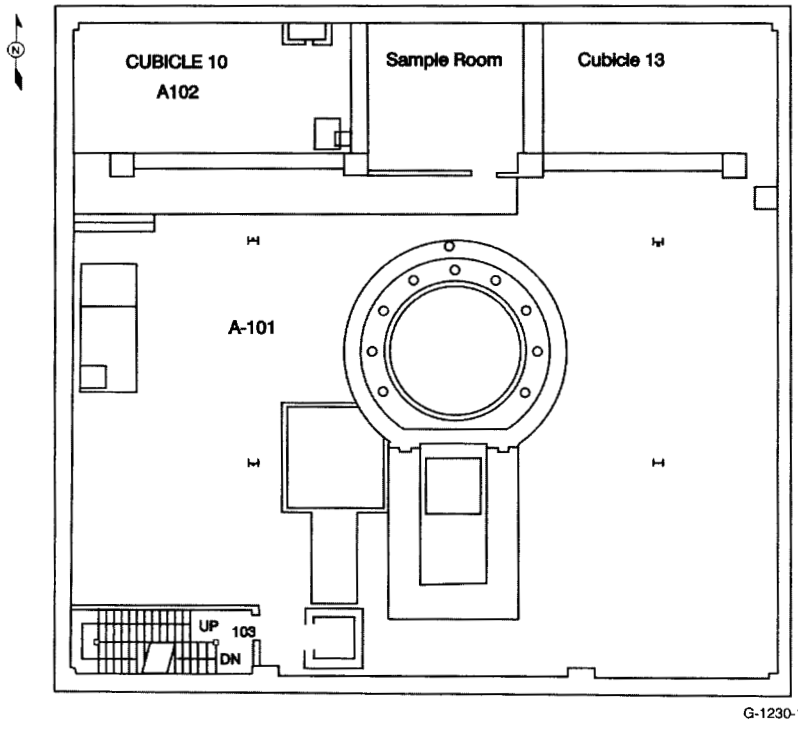

Figure 4. Power Burst Facility reactor building (PER-620) first basement showing Cubicles 10 and 13.

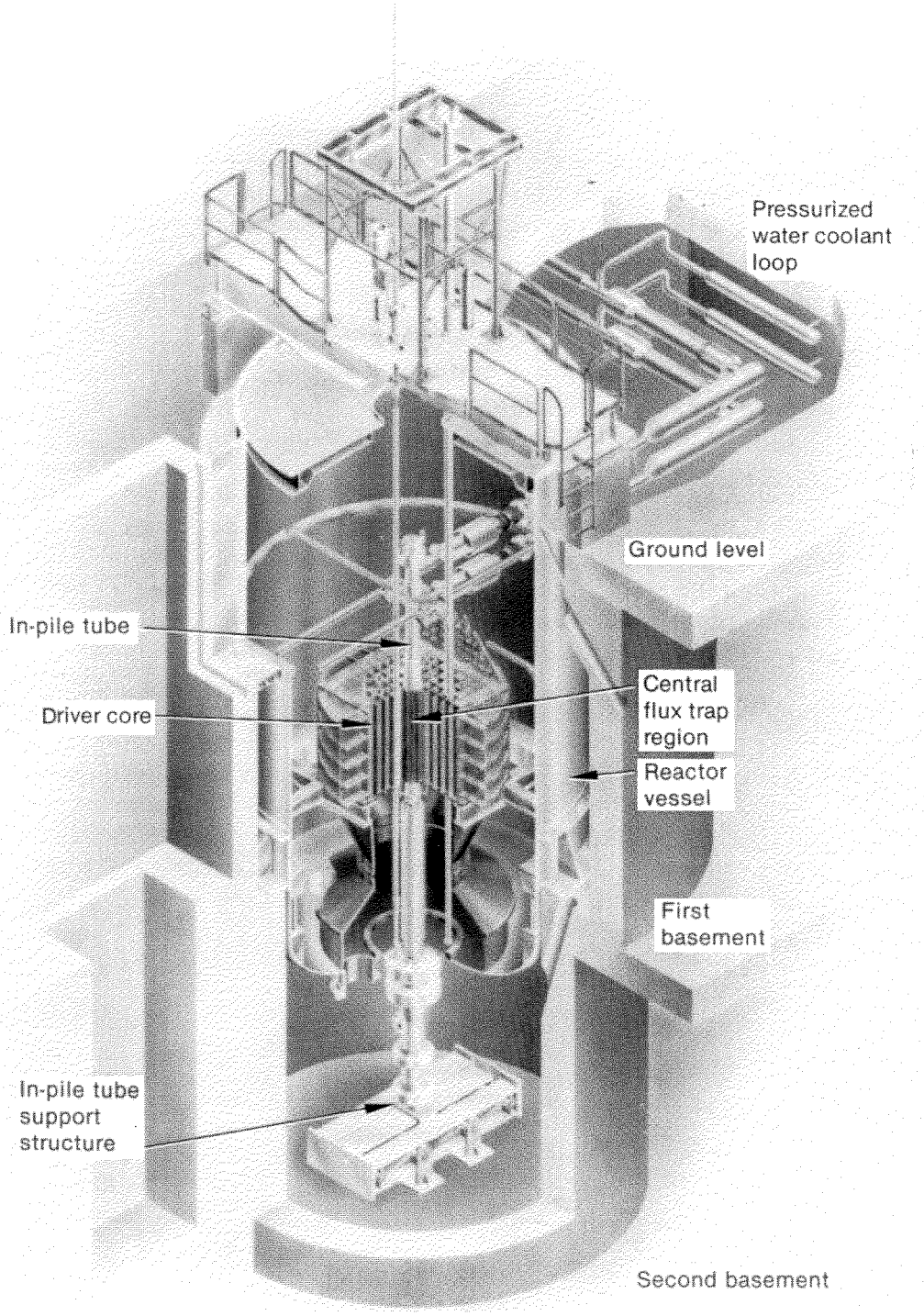

Figure 5. Power Burst Facility reactor sectional view.

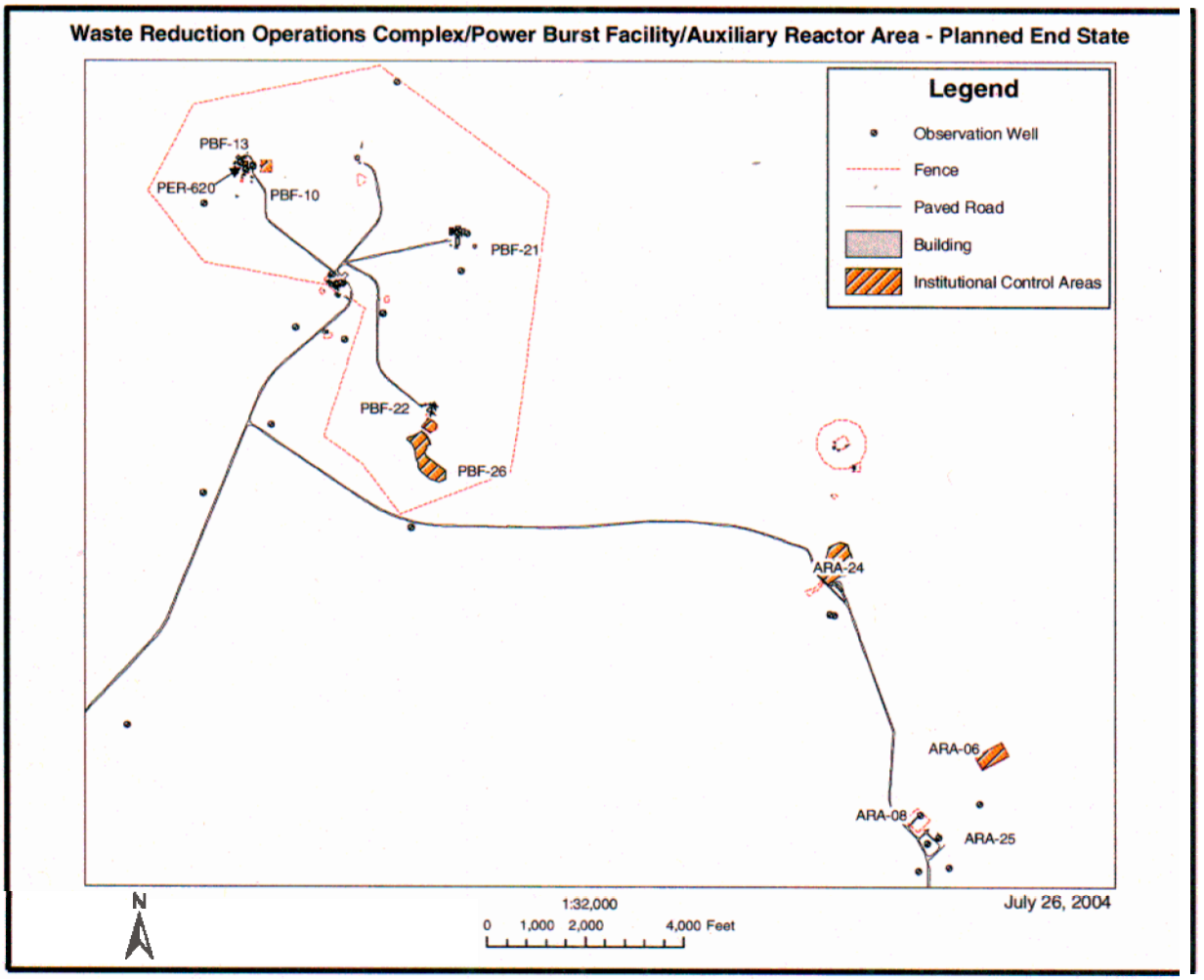

Figure 6. Locations of current and planned institutionally controlled areas at the Power Burst Facility/Auxiliary Reactor Area.

Near PER-620, long-term institutional controls are currently maintained for the following sites: ARA-06 (SL-1 burial ground), ARA-07 (ARA-II seepage pit east), ARA-08 (ARA-II seepage pit west), ARA-24 (ARA-III windblown soil), ARA-25 (soil beneath ARA-626 hot cells), PBF-10 (PBF reactor evaporation pond), PBF-12 (SPERT-I leach pond), PBF-13 (PBF area rubble pit), PBF-21 (SPERT-III large leach pond), PBF-22 (SPERT-IV leach pond), and PBF-26 (SPERT-IV lake). At the SL-1 burial grounds, radioactively contaminated debris from a steam explosion at the reactor and approximately 76.5 m^3 (1,910,000 lb) of lead was disposed of between 1961 and 1962 (DOE-ID 1999). A permanent, intrusion-resistant engineered cover is present for the SL-1 burial ground (ARA-06), since the buried debris would require isolation for a minimum of 400 years. All of the alternatives considered for this proposed removal action would be consistent with the remedial action objectives for soil sites and the future land use assumptions in the *Record of Decision Power Burst Facility and Auxiliary Reactor Area, Operable Unit 5-12* (DOE-ID 2000), including those alternatives that would leave waste in place (Alternatives 1, 2, and 4).

In June 2002, during routine gauging of an underground heating fuel storage tank located adjacent to the PBF reactor building, a decrease in the product level suggested that the tank (PER-722) might have

released fuel to the subsurface. Further investigation confirmed that heating oil was released from the tank to the subsurface. The remaining heating fuel product was removed from the tank, but the tank remains in place. Characterization studies, including the installation of borings and a monitoring well completed in the Snake River Plain Aquifer, demonstrated that the aquifer is not impacted by the release (EDF-4697). The empty tank has been filled with grout. The Idaho Department of Environmental Quality (DEQ) is requiring groundwater monitoring for a minimum of 3 years.

Under the Voluntary Consent Order (VCO) NEW-PBF-001 Action Plan, the INEEL has already characterized a total of 44 items considered Hazardous Waste Management Act/Resource Conservation and Recovery Act (HWMA/RCRA) waste at the time of the signing of the VCO. Of that total, 38 items were characterized as nonhazardous. Materials removed from the reactor building under the VCO Program include approximately 38,000 lb of lead, two panel-mounted air-conditioning units, and oil from two pump systems. The only potential HWMA/RCRA waste remaining in place that will be addressed under the alternatives for this removal action are the additional shielding lead and 147 lb of cadmium sheeting associated with the Fission Product Detection System located in Cubicle 13.

2.3 Current Closure/Cleanup Activities at the Power Burst Facility

The following sections describe cleanup and closure activities currently underway in the PER-620 buildings. These activities are outside the scope of this EE/CA and are expected to be completed by October 1, 2004. The sources were not included in the inventory for risk analysis. These activities do not impact the alternatives presented in the report.

2.3.1 Canal Deactivation Project

The PBF reactor building contains the reactor vessel, fuel storage canal, associated reactor equipment, offices, and utility rooms. Removal of the stored reactor fuel from the canal was completed in September 2003. Deactivation of the canal began in October 2003. Ongoing Canal Deactivation Project activities generally consist of removing materials and equipment from the fuel storage canal and placing the canal in a stable, low-risk condition. Deactivation includes the removal of activated fuel canisters, activated stainless-steel shim and reflector rods, aluminum filler rods, fuel rod storage racks, ion and fission chambers, a seismic support system for racks, fixed equipment, a plutonium-beryllium reactor startup source, canal water, corrosion coupons, sediment, and debris. All liquid-bearing systems would be isolated. Divers have entered the canal to seal weld the canal gate into place to isolate the reactor from the canal. In addition to installing the canal gate, the divers have removed and cleaned loose contamination from the walls and floor of the canal and applied a fixative to the canal walls and floors. The water was cleaned by filtering, and it was sent to an evaporation tank for evaporation. Canal Deactivation Project activities are expected to be completed by the end of summer 2004.

2.3.2 Initial Decommissioning Activities

Subsequent to the PBF Canal Deactivation Project, initial decommissioning activities have begun to support and facilitate the final decommissioning of PER-620. These activities include:

1. Removal of debris throughout the PBF reactor building (PER-620). Debris is defined as low-level and nonradioactive materials that include, but are not limited to, the following: tools, equipment, buckets, glassware, gas cylinders, books/manuals, and other items to be disposed of as low-level waste, industrial waste, or excess.

2. Removal of recyclable and hazardous materials in preparation for demolition of PER-620. Hazardous materials include acids, bases, some metals (i.e., lead and silver containing electrical components), polychlorinated biphenyl (PCB) -containing capacitors and ballasts, fluorescent bulbs, and other equipment and materials discovered.

3. Removal of systems and components from various aboveground rooms of PER-620. This activity includes removal of electrical cabinets, hoods, sinks, ion exchange columns, mixing tanks, and counters.

4. Transfer and disposition of radioactive and nonradioactive liquids (i.e., water and oil) from systems and components associated with PER-620. NOTE: Although disposition of these liquids is planned to be completed prior to the issuance of an Action Memorandum for the PER-620 decommissioning, if the activities are not completed prior to issuance of the Action Memorandum, and the water meets the facility's waste acceptance criteria, some of the water may be dispositioned at the INEEL CERCLA Disposal Facility (ICDF) evaporation ponds under the scope of the proposed removal action.

2.4 Extent of Contamination and Remaining Inventories

There are no known releases of contaminants from the PBF reactor building to the underlying soil. Known releases from associated systems have been evaluated and remediated, as necessary. The only known releases to the soil beneath PER-620 are the aforementioned petroleum release from an underground storage tank (PER-722) located adjacent to the PBF reactor building and releases from the warm waste and corrosive waste injection wells, previously addressed under the *Federal Facility Agreement and Consent Order for the Idaho National Engineering Laboratory* (DOE-ID 1991). Groundwater monitoring and institutional controls are the only remaining required actions related to that release. The PER-722 release is discussed in Section 2.2.

Much of this section is extracted from EDF-4869, "Groundwater Pathway Risk Assessment for the PBF Closure." Evaluations of the inventories are presented in EDF-4697, "Radiological Characterization of the PBF Reactor for Disposal," for radionuclides and EDF-4943, "Nonradiological Inventory of Materials and Components in Subgrade Basement Levels/Areas of the Power Burst Facility Reactor Building (PER-620)," for nonradionuclides. A summary of the results of these EDFs, as well as summary tables for the inventories, is presented in the following two subsections.

2.4.1 Remaining Radionuclide Inventory

A recent EDF entitled "Radiological Characterization of the PBF Reactor for Disposal" (EDF-4697) describes the evaluation of the PBF reactor building (PER-620) for activities of selected radionuclides. The analysis considered activated structures remaining in the reactor vessel; the contents of tanks and piping systems within the facility; surface contamination on the floors, walls, and ceilings of the contaminated building rooms; and selected other components. The estimated inventory is shown in Table 1. More detailed information on the radiological inventory is presented in EDF-4697 and is summarized in the following paragraphs.

Table 1. Summary of radioisotope contributions from all sources in the Power Burst Facility (in Curies [Ci]).

Isotope	Core Internals	Piping and Tanks—Internal	Piping and Tanks—External	Building and Cubicle Walls—Total	Cubicle 10 Resin Beds	Cubicle 10 Hot Spots	Cubicle 13 Hot Spots	Second Basement Resin Bed Pair	Hot/Warm Waste Room Resin Bed	All Sources (Total)
H-3	9.02E-01	—	—	—	0.00E+00	—	—	0.00E+00	0.00E+00	9.02E-01
Be-10	3.90E-06	—	—	—	8.04E-11	—	—	2.97E-11	1.01E-11	3.90E-06
C-14	1.26E-02	—	—	—	6.74E-09	—	—	1.45E-07	4.92E-08	1.26E-02
Cl-36	2.65E-04	—	—	—	0.00E+00	—	—	2.63E-09	8.94E-10	2.65E-04
Mn-54	1.28E-05	—	—	—	0.00E+00	—	—	1.12E-09	3.82E-10	1.28E-05
Ni-59	4.74E-01	—	—	—	0.00E+00	—	—	8.86E-07	3.01E-07	4.74E-01
Co-60	2.64E+01	—	—	—	0.00E+00	—	—	3.40E-04	1.16E-04	2.64E+01
Ni-63	5.00E+01	—	—	—	0.00E+00	—	—	2.50E-02	8.50E-03	5.00E+01
Zn-65	1.43E-09	—	—	—	0.00E+00	—	—	4.65E-07	1.58E-07	6.25E-07
Sr-90	2.44E-06	7.30E-02	1.19E-02	7.29E-02	1.24E+00	8.84E-06	9.99E-03	5.76E-03	1.96E-03	1.42E+00
Nb-94	1.23E-01	—	—	—	3.21E-11	—	—	2.25E-10	7.66E-11	1.23E-01
Tc-99	1.34E-04	—	—	—	2.36E-05	—	—	1.60E-02	5.43E-03	2.15E-02
Ru-103	0.00E+00	—	—	—	0.00E+00	—	—	0.00E+00	0.00E+00	0.00E+00
Ru-106	9.17E-12	—	—	—	3.18E-09	—	—	1.09E-07	3.69E-08	1.49E-07
Ag-108m	8.61E-04	—	—	—	1.25E-10	—	—	1.82E-14	6.18E-15	8.61E-04
Ag-110m	2.67E-09	—	—	—	1.47E-13	—	—	4.03E-05	1.37E-05	5.39E-05
Sb-125	4.03E-04	—	—	—	9.03E-04	—	—	1.73E-04	5.88E-05	1.54E-03
I-129	2.20E-11	—	—	—	3.28E-06	—	—	1.94E-03	6.61E-04	2.61E-03
Cs-134	7.72E-05	—	—	—	2.51E-04	—	—	2.22E-03	7.55E-04	3.30E-03
Cs-137	2.81E-06	8.29E+00	1.36E+00	8.28E+00	7.13E+00	1.01E-03	1.14E+00	6.11E-01	2.08E-01	2.70E+01
Ce-144	9.13E-13	—	—	—	3.22E-11	—	—	1.25E-08	4.24E-09	1.67E-08
Eu-152	1.01E-02	—	—	—	5.04E-04	—	—	4.86E-04	1.65E-04	1.13E-02
Eu-154	8.95E-04	—	—	—	6.82E-03	—	—	5.21E-05	1.77E-05	7.78E-03
Pb-210	9.28E-13	—	—	—	2.59E-09	—	—	1.96E-08	6.67E-09	2.89E-08

Table 1. (continued).

Isotope	Core Internals	Piping and Tanks—Internal	Piping and Tanks—External	Building and Cubicle Walls—Total	Cubicle 10 Resin Beds	Cubicle 10 Hot Spots	Cubicle 13 Hot Spots	Second Basement Resin Bed Pair	Hot/Warm Waste Room Resin Bed	All Sources (Total)
Ra-226	3.50E-12	—	—	—	2.62E-09	—	—	9.26E-08	3.15E-08	1.27E-07
Ac-227	4.55E-10	—	—	—	2.01E-08	—	—	5.69E-07	1.93E-07	7.83E-07
Th-228	2.70E-07	—	—	—	3.36E-09	—	—	1.34E-08	4.56E-09	2.91E-07
Th-229	3.27E-10	—	—	—	1.45E-13	—	—	7.48E-13	2.54E-13	3.28E-10
Th-230	5.02E-10	—	—	—	1.51E-07	—	—	1.25E-07	4.25E-08	3.19E-07
Th-232	2.79E-07	—	—	—	8.56E-16	—	—	6.08E-15	2.07E-15	2.79E-07
Pa-231	4.55E-10	—	—	—	1.91E-08	—	—	1.86E-06	6.33E-07	2.51E-06
U-232	5.12E-10	—	—	—	4.78E-09	—	—	1.31E-08	4.44E-09	2.28E-08
U-233	1.50E-07	—	—	—	6.43E-11	—	—	3.26E-10	1.11E-10	1.51E-07
U-234	1.72E-06	—	—	—	8.11E-04	—	—	5.34E-06	1.82E-06	8.20E-04
U-235	7.82E-08	—	—	—	3.66E-05	—	—	9.72E-07	3.30E-07	3.80E-05
U-236	3.74E-11	—	—	—	7.77E-07	—	—	5.03E-06	1.71E-06	7.52E-06
U-238	1.70E-06	—	—	—	1.06E-05	—	—	2.85E-05	9.68E-06	5.04E-05
Np-237	2.53E-11	—	—	—	1.18E-07	—	—	4.67E-08	1.59E-08	1.80E-07
Pu-238	4.09E-10	—	—	—	7.58E-05	—	—	4.16E-06	1.42E-06	8.14E-05
Pu-239	4.49E-07	—	—	—	3.79E-04	—	—	1.91E-05	6.49E-06	4.05E-04
Pu-240	3.15E-09	—	—	—	1.10E-04	—	—	1.91E-05	6.49E-06	1.36E-04
Pu-241	4.01E-09	—	—	—	1.45E-03	—	—	5.13E-09	1.74E-09	1.45E-03
Pu-242	3.12E-16	—	—	—	8.47E-09	—	—	2.62E-17	8.93E-18	8.47E-09
Pu-244	1.08E-27	—	—	—	1.84E-17	—	—	2.55E-33	8.67E-34	1.84E-17
Am-241	2.25E-10	—	—	—	9.58E-03	—	—	5.90E-06	2.01E-06	9.59E-03
Am-243	1.36E-17	—	—	—	4.32E-07	—	—	3.73E-20	1.27E-20	4.32E-07
Cm-243	1.81E-17	—	—	—	2.34E-07	—	—	0.00E+00	0.00E+00	2.34E-07
Cm-244	4.65E-18	—	—	—	1.25E-06	—	—	2.83E-23	9.64E-24	1.25E-06

Table 1. (continued).

Isotope	Core Internals	Piping and Tanks—Internal	Piping and Tanks—External	Building and Cubicle Walls—Total	Cubicle 10 Resin Beds	Cubicle 10 Hot Spots	Cubicle 13 Hot Spots	Second Basement Resin Bed Pair	Hot/Warm Waste Room Resin Bed	All Sources (Total)
Cm-245	5.51E-24	—	—	—	4.92E-11	—	—	0.00E+00	0.00E+00	4.92E-11
Cm-246	2.44E-27	—	—	—	9.41E-13	—	—	0.00E+00	0.00E+00	9.41E-13
Cm-247	3.65E-35	—	—	—	2.67E-19	—	—	0.00E+00	0.00E+00	2.67E-19
Cm-248	2.54E-37	—	—	—	5.89E-20	—	—	0.00E+00	0.00E+00	5.89E-20
Totals	7.79E+01	8.36E+00	1.37E+00	8.35E+00	8.39E+00	1.01E-03	1.14E+00	6.63E-01	2.25E-01	1.064E+02

The methods of evaluation included detailed computer modeling using MCNP4C, ORIGEN2, SCDAP, and CORSOR computer codes to evaluate activation of structures in and around the core, test fuel rod activation and fissioning, and release actinides and fission fragments by failed fuel rods. The contents of tanks and piping systems, and contamination of building surfaces, were evaluated by estimating volumes and surface areas, studying measurements of external gamma energy and gamma spectra, and using the MICROSHIELD computer code to assess inventory. Comparison also was made with historical data and previous sample data.

In EDF-4697, it was concluded that the overall activity in PER-620 is approximately 106 Ci, consisting of:

- 78 Ci embedded in activated structures, the IPT, and reactor vessel
- 17 Ci in tanks and piping systems, including the resin beds
- 11 Ci on exposed surfaces of the various rooms and cubicles.

The level of uncertainty in this analysis was about 50%. None of the materials would be classified as transuranic waste. The 50% uncertainty of the radiological inventory is based on a combination of factors identified in Section 7 of EDF-4697, "Radiological Characterization of the PBF Reactor for Disposal." These factors include detector measurement accuracy, surface area and volume estimations, alloy composition uncertainty, and analysis code uncertainty.

Most of the radioisotope inventory is embedded in the activated structures inside the reactor vessel. Notable among these is the IPT that housed the experiments. It was located along the center line of the core and had an estimated activity of 56.2 Ci. Of that, 43 Ci is Ni-63 and 12.6 Ci is Co-60. The rest of the reactor structures contain an aggregate of 21.8 Ci for a total of 78.0 Ci in the activated material in the reactor vessel. The structures other than the IPT contain activation products, mostly from Type 304 stainless steel. The greatest of these is 13.9 Ci of Co-60. The next most significant part of the radionuclide inventory is 8.45 Ci residing in resin beds located in Cubicle 10. The resins beds were used to clean the experiment coolant loop, removing the fission fragments and actinides lost to the loop coolant when test rods failed. Most of the radionuclide inventory in the resin beds results from Cs-137, but isotopes with long half-lives also are present. Pre-filters and strainers prevented particles and fragmented pieces of the test rods from entering the resin column and passing through the system.

Activity in the remainder of the piping and tank systems is 8.44 Ci. Least in significance is the contamination on exposed surfaces of structures. Aggregate surface contamination on walls and pipe external surfaces accounts for 11 Ci. It is effectively all Cs-137. The 50% uncertainty of the radiological inventory is based on a combination of factors identified in Section 7 of EDF-4697, "Radiological Characterization of the PBF Reactor for Disposal."

2.4.2 Remaining Nonradionuclide Inventory

The nonradionuclide inventory was estimated and the results are documented in EDF-4943, "Nonradiological Inventory of Materials and Components in Subgrade Basement Levels/Areas of the Power Burst Facility Reactor Building (PER-620)." The inventory estimates are listed in Table 2.

Table 2. Power Burst Facility reactor building (PER-620) nonradionuclide estimated inventory.

Potential Contaminant	(kg)	(lb)
Aluminum	2,041	4,490
Boron	164	361
Cadmium	67	147
Chromium[a]	21,750	47,850
Lead	146,637	322,600
Manganese[a]	2,172	4,778
Nickel[a]	11,070	24,350
Selenium	0.03	0.07
Uranium (combined U-238 and U-235 isotopes)[b]	1.109	2.44
Zinc	454	999

a. Chromium, manganese, and nickel are associated with stainless-steel piping, tanks, and other materials.
b. The estimated inventory for metallic uranium is about three times as large as the combined U-235 and U-238 inventory estimates.

The "Nonradiological Inventory of Materials and Components in Subgrade Basement Levels/Areas of the Power Burst Facility Reactor Building (PER-620)" (EDF-4943) presents the nonradiological inventory estimated for the building substructure after initial decommissioning of the facility has been completed. The estimated inventory does not include the aboveground portion of the facility, as the intent of the study was to estimate the inventory that would be left in place and stabilized under one of the alternatives considered for this removal action. The EDF contains a general description of nonradiological items that could pose a risk to human health and the environment, their location and use in the facility, physical form, and shape. The inventory for these materials and components is based either on calculations or has been approximated. Calculations were based on discussions with PBF operators, review of drawings and photographs, and evaluation of other support documentation.

With the exception of 322,000 lb of lead and 147 lb of cadmium-containing plates associated with the Fission Product Detection System in Cubicle 13, it is expected that there would be no materials remaining in the structures before initiation of the removal action that might become hazardous waste. Actions have been completed under the VCO to characterize suspected hazardous materials and systems and to remove all identified hazardous materials with the exception of material in use as shielding.

Asbestos was used in utility piping insulation (often referred to as thermal system insulation) in the two basement levels: on piping within the process areas, piping in Loop Cubicles 10 and 13, and in the knockout drum room, annulus, and other subgrade basement areas. The asbestos used in these areas is friable asbestos, as defined in 40 *Code of Federal Regulations* (CFR) 61, "National Emissions Standards for Hazardous Air Pollutants." The total amount of friable asbestos in the basement areas includes 969 linear ft of pipe insulation and mudded joints, 415 ft^2 of tank insulation, and 185 ft^2 of fire doors. Nonfriable asbestos includes 24 ft^2 of transite, 5 linear ft of caulking, and 16 ft^2 of countertops. Asbestos is present in the abovegrade structure and would be removed before demolition in all of the alternatives considered for this removal action.

Water remaining in lines and tanks associated with the PBF reactor building has mostly been characterized through laboratory analysis and through process knowledge (INEEL 2003a; Daley 1996; Scott 1996; Chinich 1987). Further characterization of less than fully characterized water is underway and would continue under the removal action as the water in vessels and lines becomes accessible. None of the water has been determined to hold the characteristics of a hazardous waste, once no longer used for shielding, nor have any of the waters been determined to contain listed constituents. Once removed, the water would be categorized as a low-level radioactive waste.

2.5 Risk Assessment

A risk assessment has been prepared that utilizes the results of the radiological and nonradiological characterization evaluations just described; it is presented in EDF-4869, "Groundwater Pathway Risk Assessment for the PBF Closure." This section is extracted from EDF-4869. The primary objective of the streamlined risk assessment is to show whether leaving the currently estimated contaminant inventories in place would meet the removal action objectives (Section 3.1). This approach taken was to evaluate a worst-case scenario, one in which the maximum mass of contaminants would be left in place. If the worst case could be shown to be protective of the groundwater pathway, then it could be assumed that other alternatives also would be protective. The intent was not to prepare risk assessments of each alternative for comparative analysis.

For purposes of the risk assessment, it was assumed that the PBF structure above grade would be removed and everything below grade would be left in place. The remaining PBF structure and contents below grade would be stabilized either by filling the void space with soil or grout. The streamlined risk assessment consisted of a screening-level evaluation, assuming that soil would be used to stabilize the PBF structure and that the contents would be left in place. Hence, the grouting described in Alternatives 1, 2, and 3 in later sections of this report would provide an additional degree of protection.

The inventory for the PBF consisted of 52 individual radionuclides and 10 nonradionuclides. The large number of contaminants presented difficulties in estimating future dose impacts and diluted resources from those nuclides that are most important. Therefore, contaminant screening was performed to reduce the number of contaminants to a manageable level and focus attention on those contaminants that are most important. Contaminant screening is discussed below. As shown below, the estimated inventories at PBF are sufficiently small that all of the 62 contaminants were screened out in the screening process and no detailed analysis was necessary.

Contaminant screening was performed in three phases:

- Phase I screening was only used for radionuclide screening. It used screening factors developed by the National Council on Radiation Protection (NCRP) (NCRP 1996) together with the estimated contaminant inventory.

- Phase II screening used a simple and conservative application of the GWSCREEN (Rood 1994) model to calculate a screening dose, risk, or concentration, based on the contaminant radionuclide or nonradionuclide inventory. The GWSCREEN application considers dilution, dispersion, and unsaturated transit time, whereas the NCRP does not. The Phase II screening application of GWSCREEN is based on the Track 2 screening approach used in the CERCLA process at the INEEL (DOE-ID 1994).

- Phase III screening is the same as Phase II screening except that contaminant solubility limits were included in the calculations. The contaminant releases from a source are sometimes solubility limited rather than K_d limited. For the PBF Project, the solubilities of chromium and lead are a major factor in the estimated transport through the vadose zone and concentrations in the aquifer.

Contaminants that are not screened would be carried forward for a more realistic analysis of their potential for risk by contamination of the aquifer.

2.5.1 Phase I (Radionuclide Only) Screening

The NCRP provides a series of simple screening techniques and factors that can be used to demonstrate compliance with environmental standards or other administratively set reference levels for releases of radionuclides to the atmosphere, surface water, or groundwater. The screening factors applicable to groundwater exposure consider leaching and subsequent dilution of radionuclides in groundwater from a generic waste site.

Of the 52 nuclides for which inventories are reported in Section 2.4 and repeated in Table 3, 38 nuclides were screened in Phase I leaving 14 nuclides for further evaluation. In addition, because Np-237 is a daughter product of Am-241, Np-237 together with the Am-241 decayed to Np-237 is retained for further evaluation. These 15 nuclides (shown in bold in Table 3) were then put through the Phase II screening.

Table 3. Phase I screening results for radionuclides using the National Council on Radiation Protection creening factors.

Radionuclide	Radioactive Half-Life (years)	PBF Inventory (Ci)	PBF Inventory (Bq)	Groundwater Ingestion NCRP Screening Factor (Sv/Bq)	Screening Dose (Sv)	Is Screening Dose <1 mrem? (<1 × 10^{-5} Sv?)
Ac-227	2.18E+01	6.18E-07	2.29E+04	8.1E-12	1.9E-07	Yes
Ag-108m	1.27E+02	8.62E-04	3.19E+07	4.2E-14	1.3E-06	Yes
Ag-110m	6.84E-01	2.67E-09	9.88E+01	5.2E-15	5.1E-13	Yes
Am-241[a]	**4.32E+02**	**9.59E-03**	**3.55E+08**	**5.9E-13**	**2.1E-04**	**No**
Am-243	7.38E+03	4.32E-07	1.60E+04	6.0E-13	9.6E-09	Yes
Be-10	1.60E+06	3.90E-06	1.44E+05	1.4E-14	2.0E-09	Yes
C-14	**5.73E+03**	**1.26E-02**	**4.66E+08**	**1.6E-13**	**7.5E-05**	**No**
Ce-144	7.78E-01	1.21E-08	4.48E+02	3.6E-15	1.6E-12	Yes
Cl-36	3.01E+05	2.66E-04	9.84E+06	8.3E-13	8.2E-06	Yes
Cm-243	2.85E+01	2.34E-07	8.66E+03	1.5E-13	1.3E-09	Yes
Cm-244	1.81E+01	1.25E-06	4.63E+04	1.1E-13	5.1E-09	Yes
Cm-245	8.50E+03	4.92E-11	1.82E+00	5.1E-13	9.3E-13	Yes
Cm-246	4.75E+03	9.41E-13	3.48E-02	2.9E-13	1.0E-14	Yes
Cm-247	1.56E+07	2.67E-19	9.88E-09	3.0E-13	3.0E-21	Yes
Cm-248	3.39E+05	5.89E-20	2.18E-09	1.1E-12	2.4E-21	Yes
Co-60	**5.27E+00**	**2.65E+01**	**9.81E+11**	**5.8E-14**	**5.7E-02**	**No**

Table 3. (continued).

Radionuclide	Radioactive Half-Life (years)	PBF Inventory (Ci)	PBF Inventory (Bq)	Groundwater Ingestion NCRP Screening Factor (Sv/Bq)	Screening Dose (Sv)	Is Screening Dose <1 mrem? (<1 × 10^{-5} Sv?)
Cs-134	2.06E+00	2.37E-03	8.77E+07	4.2E-15	3.7E-07	Yes
Cs-137	**3.02E+01**	**2.80E+01**	**1.04E+12**	**7.7E-14**	**8.0E-02**	**No**
Eu-152	1.36E+01	1.11E-02	4.11E+08	9.1E-15	3.7E-06	Yes
Eu-154	8.80E+00	7.77E-03	2.87E+08	1.1E-14	3.2E-06	Yes
H-3	**1.23E+01**	**9.03E-01**	**3.34E+10**	**5.9E-14**	**2.0E-03**	**No**
I-129	**1.57E+07**	**3.51E-06**	**1.30E+05**	**1.9E-10**	**2.5E-05**	**No**
Mn-54	8.56E-01	1.28E-05	4.74E+05	3.8E-15	1.8E-09	Yes
Nb-94	**2.03E+04**	**1.23E-01**	**4.55E+09**	**2.7E-14**	**1.2E-04**	**No**
Ni-59	7.60E+04	4.74E-01	1.75E+10	3.2E-16	5.6E-06	Yes
Ni-63	**1.00E+02**	**5.00E+01**	**1.85E+12**	**8.6E-16**	**1.6E-03**	**No**
Np-237	2.14E+06	1.63E-07	6.03E+03	2.4E-10	1.4E-06	Yes
Pa-231	3.73E+04	1.83E-06	6.77E+04	1.5E-11	1.0E-06	Yes
Pb-210	2.23E+01	2.33E-08	8.62E+02	5.4E-12	4.7E-09	Yes
Pu-238	8.78E+01	7.58E-05	2.80E+06	1.7E-12	4.8E-06	Yes
Pu-239	**2.41E+04**	**4.11E-04**	**1.52E+07**	**2.0E-12**	**3.0E-05**	**No**
Pu-240	6.57E+03	1.12E-04	4.14E+06	2.0E-12	8.3E-06	Yes
Pu-241	1.44E+01	1.45E-03	5.37E+07	6.1E-14	3.3E-06	Yes
Pu-242	3.76E+05	8.47E-09	3.13E+02	1.9E-12	6.0E-10	Yes
Pu-244	8.26E+07	1.84E-17	6.81E-07	2.2E-12	1.5E-18	Yes
Ra-226	1.60E+03	1.00E-07	3.70E+03	4.6E-12	1.7E-08	Yes
Ru-103	1.08E-01	0.00E+00	0.00E+00	1.4E-20	0.0E+00	Yes
Ru-106	1.01E+00	1.09E-07	4.03E+03	6.5E-14	2.6E-10	Yes
Sb-125	2.77E+00	1.52E-03	5.62E+07	3.6E-15	2.0E-07	Yes
Sr-90	**2.86E+01**	**1.28E+00**	**4.74E+10**	**3.5E-12**	**1.7E-01**	**No**
Tc-99	**2.13E+05**	**3.02E-04**	**1.12E+07**	**3.2E-12**	**3.6E-05**	**No**
Th-228	1.91E+00	4.89E-08	1.81E+03	2.1E-15	3.8E-12	Yes
Th-229	7.34E+03	1.71E-10	6.33E+00	3.6E-13	2.3E-12	Yes
Th-230	7.70E+04	1.02E-06	3.77E+04	5.2E-13	2.0E-08	Yes
Th-232	1.40E+10	3.32E-08	1.23E+03	4.8E-13	5.9E-10	Yes
U-232	7.20E+01	1.80E-08	6.66E+02	3.3E-11	2.2E-08	Yes
U-233	1.59E+05	7.90E-08	2.92E+03	1.1E-11	3.2E-08	Yes
U-234	**2.44E+05**	**4.23E-03**	**1.57E+08**	**4.2E-12**	**6.6E-04**	**No**
U-235	**7.04E+08**	**1.92E-04**	**7.10E+06**	**1.4E-11**	**9.9E-05**	**No**

Table 3. (continued).

Radionuclide	Radioactive Half-Life (years)	PBF Inventory (Ci)	PBF Inventory (Bq)	Groundwater Ingestion NCRP Screening Factor (Sv/Bq)	Screening Dose (Sv)	Is Screening Dose <1 mrem? (<1 × 10^{-5} Sv?)
U-236	2.34E+07	5.67E-06	2.10E+05	3.4E-12	7.1E-07	Yes
U-238	**4.47E+09**	**8.64E-05**	**3.20E+06**	**1.4E-10**	**4.5E-04**	**No**
Zn-65	6.69E-01	1.43E-09	5.29E+01	2.9E-15	1.5E-13	Yes

a. The Am-241 was simulated as Np-237 in the Phase II screening, assuming all 9.59E-03 Ci of Am-241 is instantaneously decayed and becomes 1.97E-06 Ci of Np-237.
NCRP = National Council on Radiation Protection
PBF = Power Burst Facility

2.5.2 Phase II Screening

Phase II screening used a conservative implementation of the groundwater screening model GWSCREEN Version 2.5 (Rood 1994) to calculate groundwater concentrations and ingestion doses for nuclides that were not screened in Phase I screening and the nonradionuclides. The GWSCREEN model was developed to address CERCLA sites at the INEEL. The code, coupled with a set of default parameter values identified in the CERCLA Track 2 risk assessment process (DOE-ID 1994), provides conservative estimates of groundwater concentrations and ingestion doses at the INEEL.

The GWSCREEN conceptual model is illustrated in Figure 7. Radionuclides disposed of in the PBF are assumed to be mixed homogeneously with soil and placed in a volume represented by the volume of the PBF. One-dimensional transport in a 11.6-m-thick unsaturated zone composed of sedimentary interbeds is assumed. The receptor well is placed on the downgradient edge of the PBF. The conceptual model assumes no containment or engineered barriers. The waste is then assumed to be exposed to infiltrating water and contaminants are leached from the waste and move into the subsurface. The aquifer was assumed to be homogeneous isotropic media of infinite lateral extent and finite thickness. Concentrations are then evaluated at the downgradient edge of the source. This receptor is the point where the highest concentrations in the aquifer are computed.

The radionuclide screening criteria for Phase II were set at 1/10th of the allowable drinking water dose for beta-gamma emitters of 4 mrem/yr, as stated in 40 CFR 141, "National Primary Drinking Water Regulations." Although this standard applies only to beta-gamma-emitting nuclides and is calculated using ICRP 2 methodology (ICRP 1960), the 0.4-mrem screening criterion coupled with other conservative assumptions were believed to be stringent enough to avoid screening any nuclides of importance from the inventory. In addition, a 10^{-6} risk was used as a screening criterion. The nonradionulide screening criterion for Phase II was set at the maximum contaminant level (MCL) or applicable secondary MCL or action level since no MCL had been defined.

Input data for the GWSCREEN screening simulation (Table 4) were primarily obtained from the *Track 2 Sites: Guidance for Assessing Low Probability Hazard Sites at the INEL* document (DOE-ID 1994). The dimension of the waste disposal site, Darcy velocity[a] in the aquifer, and the sedimentary interbed thickness in the unsaturated zone are site-specific values. Nuclide-specific parameter data and the peak risk are reported in the results table (Table 5).

a. Darcy velocity—unit of measure that describes movement of water through an environmental medium.

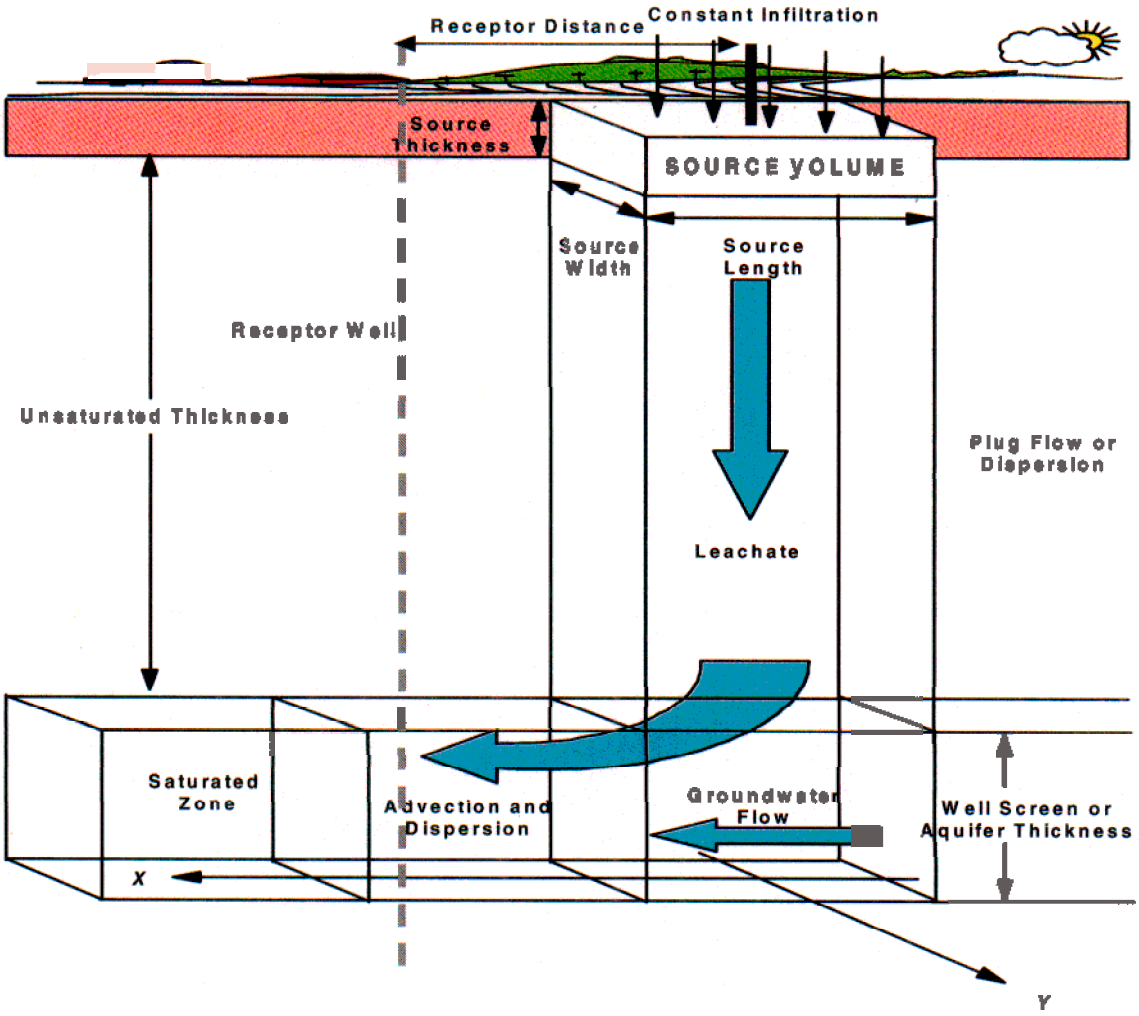

Figure 7. Conceptual model for GWSCREEN groundwater transport model.

Table 4. Parameter values for Phase II screening of the Power Burst Facility inventory.

Parameter	Value	Reference
Source		
Length parallel to groundwater flow	35 m	WAG 5 RI/FS (Holdren et al. 1999)
Width perpendicular to groundwater flow	35 m	WAG 5 RI/FS (Holdren et al. 1999)
Thickness of source	10 m	Based on the facility description.
Background percolation rate	0.1 m/yr	Track 2 document (DOE-ID 1994)
Water-filled porosity—source	0.3%	Track 2 document (DOE-ID 1994)
Bulk density—source	1.5 g/cm^3	Track 2 document (DOE-ID 1994)
Unsaturated Zone		
Cumulative vadose zone interbed thickness	10.4 m	WAG 5 RI/FS (Holdren et al. 1999), Group 3
Water-filled porosity—unsaturated zone	0.3%	Track 2 document (DOE-ID 1994)
Bulk density—unsaturated zone	1.5 g/cm^3	Assumed to be a sediment-like source
Longitudinal dispersivity	0 mL/g	Generally conservative
Aquifer		
Aquifer thickness	76 m	Track 2 document (DOE-ID 1994)
Well screen thickness[a]	15 m	Track 2 document (DOE-ID 1994)
Aquifer porosity	0.1%	Track 2 document (DOE-ID 1994)
Darcy velocity in aquifer	57 m/yr	WAG 5 RI/FS (Holdren et al. 1999) and Track 2 document (DOE-ID 1994)
Average linear velocity	570 m/yr	Darcy velocity times porosity
Longitudinal dispersivity	9 m	Track 2 document (DOE-ID 1994)
Transverse dispersivity	4 m	Track 2 document (DOE-ID 1994)
Vertical dispersivity	0.4 m	WAG 5 RI/FS (Holdren et al. 1999)
Bulk density—saturated zone	1.9 g/cm^3	Track 2 document (DOE-ID 1994)
Receptor Distance from the Center of the Source		
Parallel to groundwater flow direction	17.5 m	Based on guidance in the Track 2 document (DOE-ID 1994). Point of maximum concentration
Perpendicular to groundwater flow direction	0 m	Track 2 document (DOE-ID 1994)
Receptor Scenario		
Averaging time (radionuclides)	25,560 day	70 years, Track 2 document (DOE-ID 1994)
Water ingestion rates for receptor	2 L/d	Track 2 document (DOE-ID 1994)
Exposure frequency	365 d/yr	Track 2 document (DOE-ID 1994)
Exposure duration	30 yr	Track 2 document (DOE-ID 1994)

a. A vertically averaged solution is used in accordance with the Track 2 document (DOE-ID 2004). Thickness of the vertical section is taken to be the well screen thickness.
DOE-ID = U.S. Department of Energy Idaho Operations Office
RI/FS = remedial investigation/feasibility study
WAG = waste area group

Table 5. Radionuclide Phase II groundwater pathway screening results using Track 2 (DOE-ID 1994) assumptions.

Nuclide	Progeny	Half-life (years)	K_d (mL/g)[a]	Predicted Peak Concen. (pCi/L)	Years to Peak Concen. (years)	Ingestion DCF (rem/Ci)[b]	Peak Decay Chain Dose (mrem)[c]	Morbidity Risk Coefficient (risk/Ci)	Peak Risk
Am-241 (Np)[d]		2.14E+06	8	4.07E-05	1.28E+03	4.44E+06	1.32E-04	6.18E+01	5.51E-11
	U-233	1.59E+05	6	2.89E-07	—	2.89E+05	6.09E-08	7.18E+01	4.54E-13
	Th-229	7.34E+03	100	1.20E-09	—	4.03E+06	3.53E-09	2.24E+02	5.88E-15
Total Am-241 (Np)		—	—	—	—	—	**1.32E-04**	—	**5.55E-11**
C-14		5.73E+03	0.1	7.03E+00	4.72E+01	2.09E+03	**1.07E-02**	1.55E+00	**2.39E-07**
Co-60		5.27E+00	10	0.00E+00	NA	2.69E+04	**NA**	1.57E+01	**NA**

Table 5. (continued).

Nuclide	Progeny	Half-life (years)	K_d (mL/g)[a]	Predicted Peak Concen. (pCi/L)	Years to Peak Concen. (years)	Ingestion DCF (rem/Ci)[b]	Peak Decay Chain Dose (mrem)[c]	Morbidity Risk Coefficient (risk/Ci)	Peak Risk
Cs-137		3.02E+01	500	0.00E+00	NA	5.00E+04	**NA**	3.04E+01	**NA**
H-3		1.23E+01	0	1.29E+02	3.15E+01	6.40E+01	**6.01E-03**	5.07E-02	**1.43E-07**
I-129		1.57E+07	0	2.95E-03	3.15E+01	2.76E+05	**5.94E-04**	1.48E+02	**9.53E-09**
Nb-94		2.03E+04	100 (1)	1.22E-01	1.57E+04	7.14E+03	**6.35E-04**	7.77E+00	**2.07E-08**
Ni-63		1.00E+02	100	6.62E-46	1.56E+04	5.77E+02	**2.79E-49**	6.70E-01	**9.70E-54**
Pu-239		2.41E+04	22[e]	2.84E-03	3.47E+03	3.54E+06	7.33E-03	1.35E+02	8.37E-09
	U-235	7.04E+08	6	3.25E-08	—	2.67E+05	6.34E-09	6.96E+01	4.95E-14
	Pa-231	3.73E+04	550 (2)	1.57E-11	—	1.06E+07	1.21E-10	1.73E+02	5.94E-17
	Ac-227	2.18E+01	450 (2)	1.88E-11	—	1.48E+07	2.03E-10	2.01E+02	8.29E-17
Total Pu-239		—		—	—	—	**7.33E-03**	—	**8.37E-09**
Sr-90		2.86E+01	12 (1)	1.55E-19	1.91E+03	1.42E+05	**1.60E-20**	5.59E+01	**1.89E-25**
Tc –99		2.13E+05	0.2 (1)	1.27E-01	6.28E+01	1.46E+03	**1.36E-04**	2.75E+00	**7.67E-09**
U-234g		2.44E+05	6	1.15E-01	9.70E+02	2.83E+05	2.38E-02	7.07E+01	1.78E-07
	Th-230	7.70E+04	100	7.39E-05	—	5.48E+05	2.96E-05	9.10E+01	1.47E-10
	Ra-226	1.60E+03	100	1.36E-05	—	1.33E+06	1.32E-05	3.85E+02	1.15E-10
	Pb-210	2.23E+01	100	1.28E-05	—	7.27E+06	6.78E-05	8.81E+02	2.46E-10
Total U-234[g]		—	—	—	—	—	**2.39E-02**	—	**1.79E-07**
U-235g		7.04E+08	6	5.25E-03	9.70E+02	2.67E+05	1.02E-03	6.96E+01	7.99E-09
	Pa-231	3.73E+04	550	1.41E-06	—	1.06E+07	1.09E-05	1.73E+02	5.34E-12
	Ac-227	2.18E+01	450	1.67E-06	—	1.48E+07	1.80E-05	2.01E+02	7.34E-12
Total U-235		—	—	—	—	—	**1.05E-03**	—	**8.00E-09**
U-238		4.47E+09	6	2.36E-03	9.70E+02	2.70E+05	4.65E-04	6.40E+01	3.31E-09
	U-234	2.44E+05	6	6.47E-06	—	2.83E+05	1.34E-06	7.07E+01	1.00E-11
	Th-230	7.70E+04	100	2.08E-09	—	5.48E+05	8.30E-10	9.10E+01	4.14E-15
	Ra-226	1.60E+03	100	2.63E-10	—	1.33E+06	2.55E-10	3.85E+02	2.21E-15
	Pb-210	2.23E+01	100	2.39E-10	—	7.27E+06	1.27E-09	8.81E+02	4.61E-15
Total U-238		—	—	—	—	—	**4.66E-04**	—	**3.32E-09**

a. Unless otherwise noted, all K_d values are from the Track 2 document (DOE-ID 1994). Noted sources are (1) DOE-ID (1997b), (2) Sheppard and Thibault (1990), and (3) NCRP (1996).
b. Dose conversion factors are from EPA (1988).
c. Drinking water doses were based on ingestion of 2 L of water per day for 365 days per year.
d. Neptunium-237 was the decay chain modeled in the simulation. All Am-241 activity was converted to equivalent activity of this Np-237.
e. Plutonium K_d has been shown to be much larger than 22 mL/g, but the Track 2 value is used here for screening purposes.
DCF = dose conversion factor
DOE-ID = U.S. Department of Energy Idaho Operations Office
EPA = U.S. Environmental Protection Agency
NA = not applicable

Using the 0.4-mrem/yr and 10^{-6} risk screening criteria, all of the remaining nuclides were screened and removed from further consideration (Table 5). Note that this assessment used basically the Track 2 screening assumptions and K_d values for soil. Any closure action (such as grouting) would tend to decrease the contaminants' mobility and further decrease the predicted dose and risk in the aquifer. In addition, any soil cover that decreases the infiltration rate from 10 cm/yr (Track 2 screening value) to 1 cm/yr (undisturbed INEEL sediment estimate) would decrease the predicted peak concentration by an order of magnitude.

The chemical-specific nonradionuclide parameter data and calculated peak groundwater concentrations are reported in Table 6.

Table 6. Nonradionuclide Phase II groundwater pathway screening results using Track 2 (DOE-ID 1994) assumptions.

Contaminant	K_d (mL/g)	Concentration Limit (mg/L)	Predicted Peak Concentration (mg/L)	Years to Peak Concentration (years)	Predicted Peak <MCL?
Aluminum	250	0.05 to 0.2 (secondary MCL)	1.38E-03	3.91E+04	Yes
Boron	0[a]	1 (unregulated, but under consideration)[b]	1.38E-01	3.15E+01	Yes
Cadmium	6	0.005 (MCL)	1.83E-03	9.69E+02	Yes
Chromium	**1.2**	**0.1 (total) (MCL)**	**2.63E+00**	**2.19E+02**	**No**
Lead	**100**	**0.015 (action level)**	**2.48E-01**	**1.57E+04**	**No**
Manganese	50	0.05 (secondary MCL)	7.33E-03	7.85E+03	Yes
Nickel[c]	100	0.1 (remanded MCL)	1.87E-02	1.57E+04	Yes
Selenium	4	0.05 (MCL)	1.21E-06	6.57E+02	Yes
Uranium	6	0.03 (MCL)	3.03E-05	9.69E+02	Yes
Zinc	16	5 (secondary MCL)	4.75E-03	2.53E+03	Yes

MCL = maximum contaminant level as set by the EPA.
Secondary MCL—as set by the EPA. Nonenforceable guidelines. Regulate contaminants that could cause cosmetic or esthetic effects.
Action level—lead is regulated by a treatment technique that requires systems to control the corrosiveness of their water.
a. The boron K_d is conservatively chosen to be 0 mL/g, because no K_d value is available in the INEEL-related literature.
b. Boron is unregulated by the EPA. However, boron is being evaluated for future regulation. The state of California Department of Health Services is using 1 mg/L as an action level. Therefore, 1 mg/L is provided in this table for comparison.
c. The MCL and the MCL goal for nickel were remanded on February 9, 1995. This means that there is currently no EPA legal limit on the amount of nickel in drinking water. The remanded MCL was 0.1 mg/L, so that value was used for this screening.
EPA = U.S. Environmental Protection Agency
INEEL = Idaho National Engineering and Environmental Laboratory
MCL = maximum contaminant level

2.5.3 Phase III (Nonradionuclide Only) Screening

A primary assumption of the Track 2 screening is that the contaminants are completely soluble in water and immediately available for leaching and transport to the aquifer. In the cases of chromium and lead at PBF, this is an extremely conservative assumption. In reality, the chromium is part of the stainless steel, primarily in pipes, and the lead is in lead bricks used as activity shields in the facility. These

contaminants are not readily available for transport as assumed in the screening. The metal would have to corrode before the chromium and lead are available for transport. After corrosion, the chemical solubility rather than a simple K_d process would control the leaching of the chromium and lead. Therefore, a modification is added to this screening process incorporating the solubility of the chromium and lead into the screening process. This is still a conservative assumption, because the chromium and lead are assumed to be available for transport immediately, while in reality, corrosion is necessary for the release of chromium and lead.

Using the Track 2 screening assumptions, the predicted peak aquifer concentrations are shown in Table 7. The estimated peak concentrations are smaller than the MCL by a factor of 600 for chromium and 300 for lead. The chromium that would corrode from the stainless steel into the PBF and then be leached to the subsurface is screened from the potential contaminants of concern. In addition, lead that would corrode from lead bricks into the PBF and then be leached to the subsurface is screened from the potential contaminants of concern. Immediately following grouting, the relatively high pH of wetted concrete in contact with the encapsulated metal would result in a temporary increase in the corrosion rate. As the grout cures and the moisture content is reduced, the observed corrosion rates will decrease. Once cured, the grout will impede the flow of water and air to the metal surface reducing the observed corrosion rates below those indicated in Section 2.5.3. If a crack develops or leaching occurs, and water is allowed to flow past the encapsulated metal, the high pH of the water will increase the corrosion rates of the metals above those indicated in Section 2.5.3.

Note that in addition to the conservatism built into the Track 2 screening process, the corrosion of stainless steel and lead was assumed to be instantaneous for this evaluation. In reality, stainless steel is expected to corrode at a rate of 2E-04 mm/year (Case et al. 2000) and lead is expected to corrode at a rate of 4E-04 to 6E-04 mm/year (Reich, Leitus, and Shalev 2003). This corrosion release mechanism would further reduce the predicted peak aquifer concentration of chromium and lead.

Table 7. Nonradionuclide Phase II groundwater pathway screening results using Track 2 (DOE-ID 1994) assumptions.

Contaminant	K_d (mL/g)	Solubility Limit (mg/L)	MCL (mg/L)	Predicted Peak Concen. (mg/L)	Years to Peak Concen. (years)	Predicted Peak < MCL?
Chromium	1.2	5.20E-02	0.1 (total)	1.62E-04	210[a]	Yes
Lead	100	1.65E-01	0.015 (action level)	5.15E-04	16,000[b]	Yes

a. The chromium release is a constant for many years and results in a constant aquifer concentration in 200 years that would continue for about 3 million years at an infiltration rate of 10 cm/year.
b. The lead release is a constant for many years and results in a constant aquifer concentration in 16,000 years that would continue for about 7 million years at an infiltration rate of 10 cm/year.
MCL = maximum contaminant level

2.5.4 Summary of Screening Results

As shown in Sections 2.5.1 through 2.5.3, all contaminants were eliminated in the screening process. For the radionuclides, 38 of 52 nuclides were screened out in Phase I. Neptunium-237 was screened but was carried through to Phase II, because Am-241 was not screened and Np-237 is a decay product of Am-241. The remaining radionuclides were screened in Phase II. As noted in Section 2.4.1, the radionuclide inventory has an uncertainty of about 50%. With an increase of 50% in the radionuclide inventory, all nuclides would still be screened out. For the nonradionuclides, 8 of 10 were screened in the

Phase II screening and the remaining two (chromium and lead) were screened in the Phase III screening. When the solubility of chromium and lead was incorporated into the analysis, the chromium and lead were screened from further investigation.

A number of conservative assumptions are used in the screening analysis. The following is a list of the primary assumptions and their impact on the predicted contaminant concentrations in the aquifer.

1. The infiltration rate is assumed to be 10 cm/yr, based on the Track 2 screening assumptions. If a protective cover is placed over the PBF that reduces the infiltration rate to undisturbed background rates, water would move through the contaminated soil at a rate of 1 cm/yr. This would decrease the predicted peak concentrations in the aquifer by a factor of 10 or more.

2. If the basins are filled with grout, the contaminants would be immobilized for 500 years. At 500 years, the grouted basins would begin to fail and water would be able to move through the basins, leaching out contaminants. This would move the predicted peak concentrations out at least 500 years.

3. Soil, rather than grout, K_d values are used for the screening calculations. In general, contaminants are less mobile in a high pH environment. Therefore, if the PBF is filled with grout, the contaminant mobility would generally decrease, which would decrease the predicted contaminant concentrations in the aquifer.

4. The contaminant solubility is conservatively assumed to be infinite for the Phase I and II screening analyses. If a contaminant appears to pose a significant risk to the groundwater quality, then a reasonable solubility limit could be identified and later incorporated into the analysis.

5. Some of the contaminants are currently in the form of solid pieces of metal (such as stainless-steel pipes and lead bricks). The screening assumes that the contaminants are readily available for leaching out of PBF to the vadose zone and then transported to the aquifer. In fact, the metal must first corrode before the contaminants become available for transport. The predicted peak concentrations are conservative.

6. In the screening, water and contaminants are assumed to move straight down through the vadose zone sediments. The contaminant velocity through the sediments depends on the contaminant-specific sediment K_d. There is no retardation effect from the basalt and there is no horizontal spreading in the vadose zone.

7. For the screening, the receptor is assumed to be at the edge of the PBF. This is the location of the peak concentration in the aquifer. Any movement of the receptor would result in decreased predictions of peak concentration.

8. The peak concentrations range over many millennia. Therefore, very few of the predicted peak concentrations would be expected to occur simultaneously.

2.5.5 Summary of Risk Assessment

As discussed above, the screening calculations were done making the assumption that the basins are filled with soil (rather than grout) and the current inventory below grade is assumed to remain in the PBF. In the case of the radionuclides, the analysis shows that there are no nuclides that are expected to be transported to the aquifer at concentrations greater than a 0.4-mrem/yr dose or a 10^{-6} risk. As for the nonradionuclides, no contaminants are expected to reach the aquifer at concentrations greater than the MCL. In order to screen all of the metals, solubility-limited release was assumed for chromium in the stainless-steel pipes and lead in lead bricks used to make radiation shielding walls. Since all the

contaminants were screened out, there was no modeling performed using realistic (as opposed to conservative) parameter values to estimate realistic future concentration in the aquifer.

Based on this streamlined risk assessment, filling the PBF with either soil or grout while leaving all current source inventory in place results in predicted groundwater concentrations that meet the required performance criteria. For groundwater, the performance criteria are to prevent migration of contaminants from the PBF that would cause the Snake River Plain Aquifer to exceed a cumulative carcinogenic risk level of 1×10^{-4}, a total hazard index of one, or applicable State of Idaho groundwater quality standards in 2095 and beyond.

From a cumulative risk standpoint, this streamlined risk assessment demonstrates that leaving contaminants in place in the PBF substructure would result in an insignificant contribution to the cumulative risk at Operable Unit (OU) 5-12. The concentrations of contaminants predicted in the future in the aquifer, as a result of leaving PBF contaminants in place, are orders of magnitude below the risk-based concentrations corresponding to the remedial action objectives defined in the *Record of Decision Power Burst Facility and Auxiliary Reactor Area, Operable Unit 5-12* (DOE-ID 2000).

3. IDENTIFICATION OF REMOVAL OBJECTIVES AND SCOPE

This section identifies the removal action goals, defines the scope of work, and provides a general schedule for the activities associated with this removal action.

3.1 Removal Action Objectives

The removal action objectives for this non-time critical removal action are as follows:

- Reduce the threat to the Snake River Plain Aquifer by removing the reactor/canal water and other radioactively contaminated water in storage
- Inhibit direct exposure to radionuclide contaminants of concern remaining at the PBF reactor that would result in a total excess cancer risk greater than or equal to 1 in 10,000 for future residents and for current and future workers
- Inhibit dermal adsorption of contaminants of concern remaining at the PBF reactor that would result in a total excess cancer risk greater than or equal to 1 in 10,000 or a hazard index of 2 or greater for future residents and for current and future workers
- Prevent migration of contaminants from the PBF reactor that would cause the Snake River Plain Aquifer groundwater to exceed a cumulative carcinogenic risk level of 1×10^{-4}, a total hazard index of one, or applicable State of Idaho groundwater quality standards in 2095 and beyond.

Although the PBF was not specifically addressed in the *Record of Decision Power Burst Facility and Auxiliary Reactor Area, Operable Unit 5-12* (DOE-ID 2000), these removal action goals are consistent with the remedial action objectives for contaminated soil established in the Record of Decision. The removal action goals also are predicated on the current and future land uses established for the PBF area in the Record of Decision, which includes industrial land use until at least 2095 and possible residential land use thereafter. The PBF reactor building could be considered a newly identified CERCLA site and the results of the removal action would be evaluated in one of the 5-year reviews for OU 5-12 or as a new site in the upcoming OU 10-08 Sitewide Record of Decision. The Agencies would determine the appropriate CERCLA path for evaluation of the non-time critical removal action.

Existing institutional controls would be maintained until the selected removal action has been implemented. Because contamination would remain in the subsurface after completion of the removal action, institutional controls consisting of signs, access controls, land use restrictions, and possibly other controls would be established and maintained until discontinued based on the results of a 5-year review. Institutional controls and the 5-year reviews would be conducted under either the existing Record of Decision (DOE-ID 2000) or the upcoming OU 10-08 Record of Decision.

3.2 Determination of Future Removal Action Scope

The scope of this future removal action is limited to completion of remaining actions for the PBF reactor building (PER-620). Past and current actions for PER-620 are as described in Sections 2.3.1 and 2.3.2 and are not in the scope of this future removal action. The scope does not include D&D of other PBF structures.

4. IDENTIFICATION OF REMOVAL ACTION ALTERNATIVES

Five alternatives were identified for this possible removal action, including a no action alternative.

4.1 Overview and Common Elements

With the exception of the no action alternative, the alternatives described in Section 4.2 contain some common elements. An overview of the alternatives is provided in Figure 8. Sections 4.1.1, 4.1.2, and 4.1.3 provide discussion of common scope elements, where applicable. Darkened bars in the following chart indicate the alternative of the corresponding scope element.

4.1.1 Remove Water in Tanks and Piping (Applies to Alternatives 1–4)

Water would be removed from the PER-620 facility in two phases. Phase 1 includes removing water from the piping, vessels, and tanks. This work would be accomplished using both the existing PBF pumps and tanks along with portable pumping systems and drums. The optimum removal and disposal approach for each individual water source in the facility would be identified.

Phase 2 consists of draining the water from the reactor vessel and primary coolant piping, which would occur at the same time grout is used to fill the reactor. By alternating water removal with grout fill into the reactor vessel, and thereby maintaining the surface of the water near the top of the reactor vessel, personnel shielding would not be compromised during the grouting of the vessel.

The appropriate disposal path for each water source would be identified. Final disposal approaches for each water source would be developed. Potential disposal options include the CFA sewage treatment plant for nonradioactive water, Test Reactor Area (TRA) evaporation pond, ICDF evaporation pond, PER-706 evaporation tank, or other suitable disposal facility.

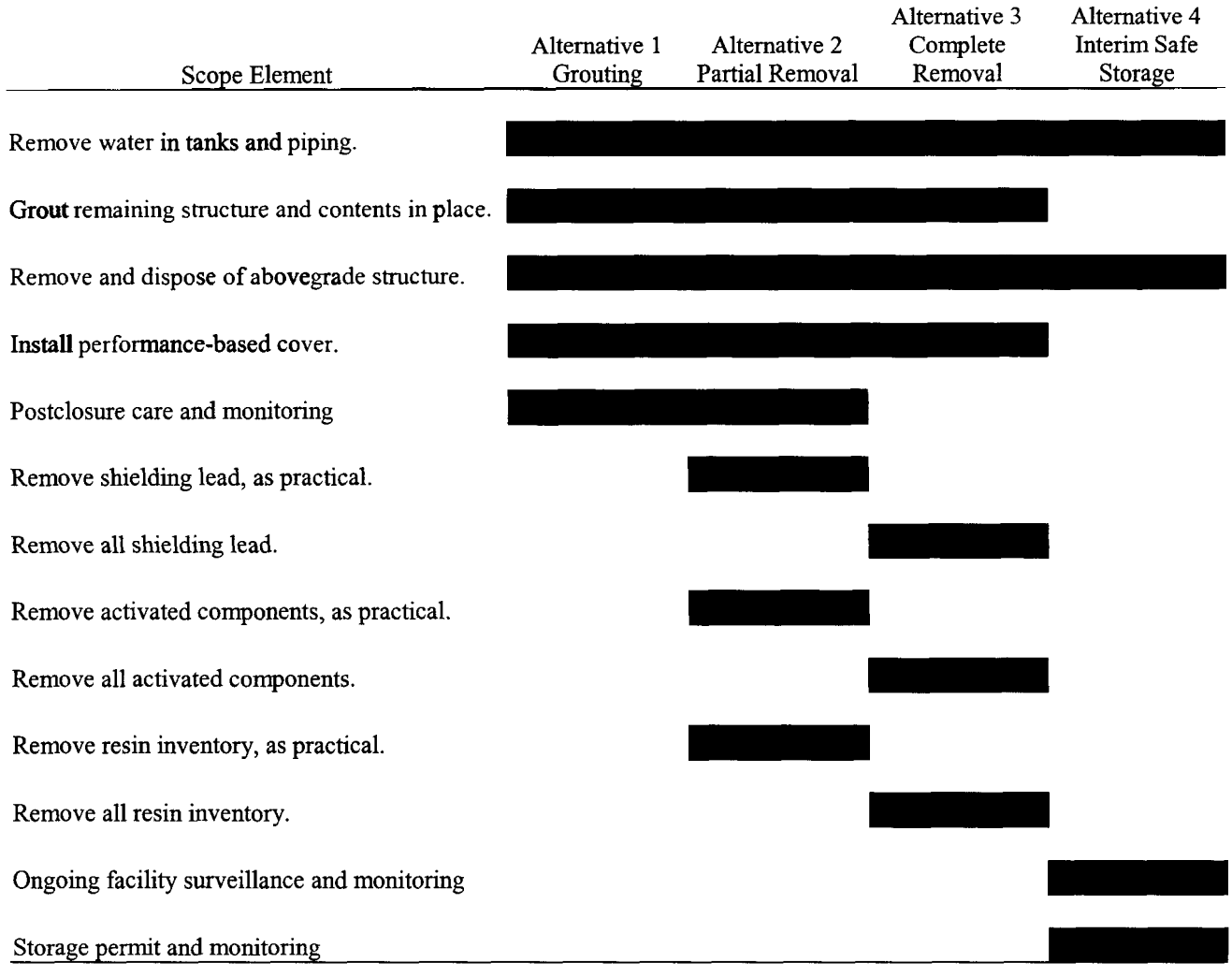

Figure 8. Overview of alternatives and common scope elements.

4.1.2 Grout Remaining Substructure and Contents in Place (Applies to Alternatives 1, 2, and 3)

Grout placement includes all work necessary to encapsulate the PER-620 rooms, cubicles, and equipment from the lowest elevation to the existing first floor at grade. The fill material would consist of a cement-based flowable grout. The substructure areas (including remaining shielding lead, resin beds, and surface contamination) would be stabilized in place.

Two grout formulations have been considered for use. One is a controlled low-strength grout, which would be used to encapsulate the belowgrade equipment, piping, and structure. The other is a higher-strength formulation for use underwater in grouting the reactor vessel. The first grout formula, which is planned for the belowgrade structures, was designed to accomplish multiple criteria: meet the compressive strength requirements due to the changing loads on the structure, minimize void space, minimize subsidence, reduce permeability, allow nominal cure times between pours, minimize heat of hydration issues, maximize flowability, and achieve reasonable costs for material. All pipes and tanks

would be drained prior to grouting. Core holes would be cut at various locations in the main floor, in the floor of the first basement, and in tanks, walls, and other areas (as necessary) to allow grout to flow freely into all areas of the PER-620 facility. A temporary ventilation system would be installed for contamination control. Cutting and capping of all process or waste lines that exit the facility below grade at the exterior wall interface would be accomplished before placement of fills in the basement areas.

The grouting activities would be accomplished in three phases with multiple lifts per phase. The first phase of the grouting begins at the primary cooling loop system, including the heat exchangers and the loop coolant system. The primary cooling loop would be filled in sequenced placements to allow isolation from the reactor vessel. The second phase of grouting includes filling the rooms, tanks, secondary cooling loop, and 10-in. or greater diameter pipes. The third phase of grouting includes the reactor vessel. To maintain shielding, the grout would be pumped into the reactor vessel with water still in the vessel, as discussed in Section 4.1.1. Displaced water would be pumped to the PER-706 evaporation tank. Void space would be minimized, as discussed in the previous paragraph, for all phases of grouting to ensure structural integrity and minimize the potential for future subsidence.

4.1.3 Remove and Dispose of Abovegrade Structure; Install Performance-Based Cover (Applies to Alternatives 1–3)

Upon completion of facility grouting, the abovegrade structures would be removed. The process for removing the abovegrade support structures would include the following tasks. Abovegrade nonstructural components would be removed and dismantled. Then the control bridge would be removed. Sizing and packaging requirements would be determined by the selected disposal path. Potential disposal pathways include the ICDF, RWMC, and CFA landfill. Waste would be managed pursuant to the requirements of the selected facility.

Once the aboveground structure has been removed, a cover designed to meet RCRA landfill cover requirements (40 CFR 264.310) would cover the grouted basements and foundation. The footprint would extend beyond the existing foundation with an appropriate slope for drainage. For the purposes of evaluation, the cover was assumed to consist of a concrete cap overlain by a soil cover. The purpose of the concrete cap is to provide an engineered seal of any small holes, penetrations, or conduits that protrude vertically from the remaining grout slab. The primary design requirement is that the permeability of the cover must be less than the permeability of the grouted mass. The footprint of the cover would extend approximately 24 ft past the original facility foundation. Soil used in the cover would be obtained from a barrow area near CFA and compacted to appropriate requirements. Seeding and planting of disturbed areas with a mixture of native grass seeds would follow. The actual cover design would be developed, with agency concurrence, as part of the Removal Action Plan.

4.1.4 Postclosure Care and Monitoring

The substantive requirements of HWMA/RCRA require long-term groundwater monitoring of one upgradient well and three downgradient wells. It is assumed this would necessitate the installation of two additional wells to supplement the existing upgradient and downgradient wells. The long-term monitoring also would require annual cover inspection and maintenance.

Since Alternatives 1 and 2 would leave residual lead and radioactive material in place following decommissioning, the Idaho Completion Project (ICP) would need to implement monitoring, maintenance, and institutional control requirements for the monolith. These requirements would fall into three general subsets: (1) cover inspection and maintenance, (2) groundwater monitoring, and (3) restrictions on future land use where the monolith is located. The DOE-ID has determined, in consultation with the DEQ and EPA, that the substantive standards of the HWMA/RCRA hazardous

waste landfill requirements would apply to the disposal of lead in the grouted subsurface structure. These requirements are applicable or relevant and appropriate requirements (ARARs) (42 USC § 9621) incorporated into the design and execution of the CERCLA non-time critical removal action and may in addition be met through obtaining a HWMA/RCRA postclosure permit from DEQ.

Although HWMA/RCRA would primarily govern the postclosure care and monitoring requirements for Alternatives 1 and 2, asbestos waste disposal requirements also would be applicable (40 CFR 61.154, "Standards for Active Waste Disposal Sites"). In addition, the decommissioning would need to be consistent with the remedial action objectives in the Record of Decision (DOE-ID 2000). Groundwater risk analysis has already confirmed that Alternatives 1 and 2 are consistent with the remedial action objectives in the Record of Decision (DOE-ID 2000) and that no adverse impacts to human health or the environment would result from the radiological or nonradiological constituents remaining in the monolith. Drinking water MCLs for the radionuclides and action levels for lead also would be ARARs. However, the groundwater modeling also has confirmed that potential contaminant migration from the monolith would not exceed the standards.

4.1.4.1 Cover Inspection and Maintenance. Annual cover inspections and a postclosure care inspection plan would be required in accordance with Idaho Administrative Procedures Act (IDAPA) 58.01.05.008 (40 CFR 264.15). Inspection logs would be required to document all inspections, deficiencies noted, and any follow-up actions recorded. Any deterioration or malfunction discovered by an inspection would require remedy.

The monolith would require maintenance in accordance with the provisions of IDAPA 58.01.05.008 (40 CFR 264.117). The INEEL would be required by IDAPA 58.01.05.008 (40 CFR 264.310) to maintain the integrity and effectiveness of the cover, including making repairs to the cover as necessary to correct the effects of settling, subsidence, erosion, or other events. Run-on and run-off controls would need to remain functional. The INEEL also would need to install, protect, and maintain surveyed benchmarks in accordance with IDAPA 58.01.05.008 (40 CFR 264.309).

4.1.4.2 Groundwater Monitoring. Routine groundwater monitoring near PER-620 is ongoing to meet CERCLA requirements established in the *Groundwater Monitoring Plan for the Waste Area Group 5 Remedial Action* (DOE-ID 2003a) and to meet DEQ Risk-Based Corrective Action requirements identified in the *Groundwater Monitoring Plan for the PER-722 Underground Storage Tank Diesel Fuel Release* (DOE-ID 2004a). Current monitoring well locations and groundwater elevations are shown in Figure 9. Additional groundwater monitoring for the grouted monolith would be required by IDAPA 58.01.05.008 (40 CFR 264, Subpart F). The *RCRA Ground-water Monitoring Draft Technical Guidance* (EPA 1992) stipulates a minimum of one upgradient well and three downgradient wells for a detection monitoring program and that the monitoring will continue for a minimum of 100 years.

The PBF-MON-A-001 well is located approximately 1,100 ft to the southwest and slightly crossgradient from PER-620, and it is of suitable construction as one of the three downgradient wells. The PBF-MON-A-005 well is located approximately 3,000 ft to the northeast of PER-620, and it is suitable for use as the upgradient well. These two wells are currently sampled annually for volatile organic compounds, total metals, and radionuclides for CERCLA purposes. A third potential monitoring point is the SPERT-1 production well located at the PBF control area. However, SPERT-1 is approximately 0.5 mi downgradient of PER-620. The only other well in the vicinity of PER-620 is PBF-1930, which is located immediately to the north of the building. Monitoring of PBF-1930 is required by the DEQ Risk-Based Corrective Action Program for benzene, toluene, ethylbenzene, total xylenes, and polycyclic aromatic hydrocarbons on a quarterly basis for a minimum of 3 years. However, PBF-1930 is a piezometer well constructed of polyvinyl chloride pipe and presumably would not be suitable as a long-term monitoring point.

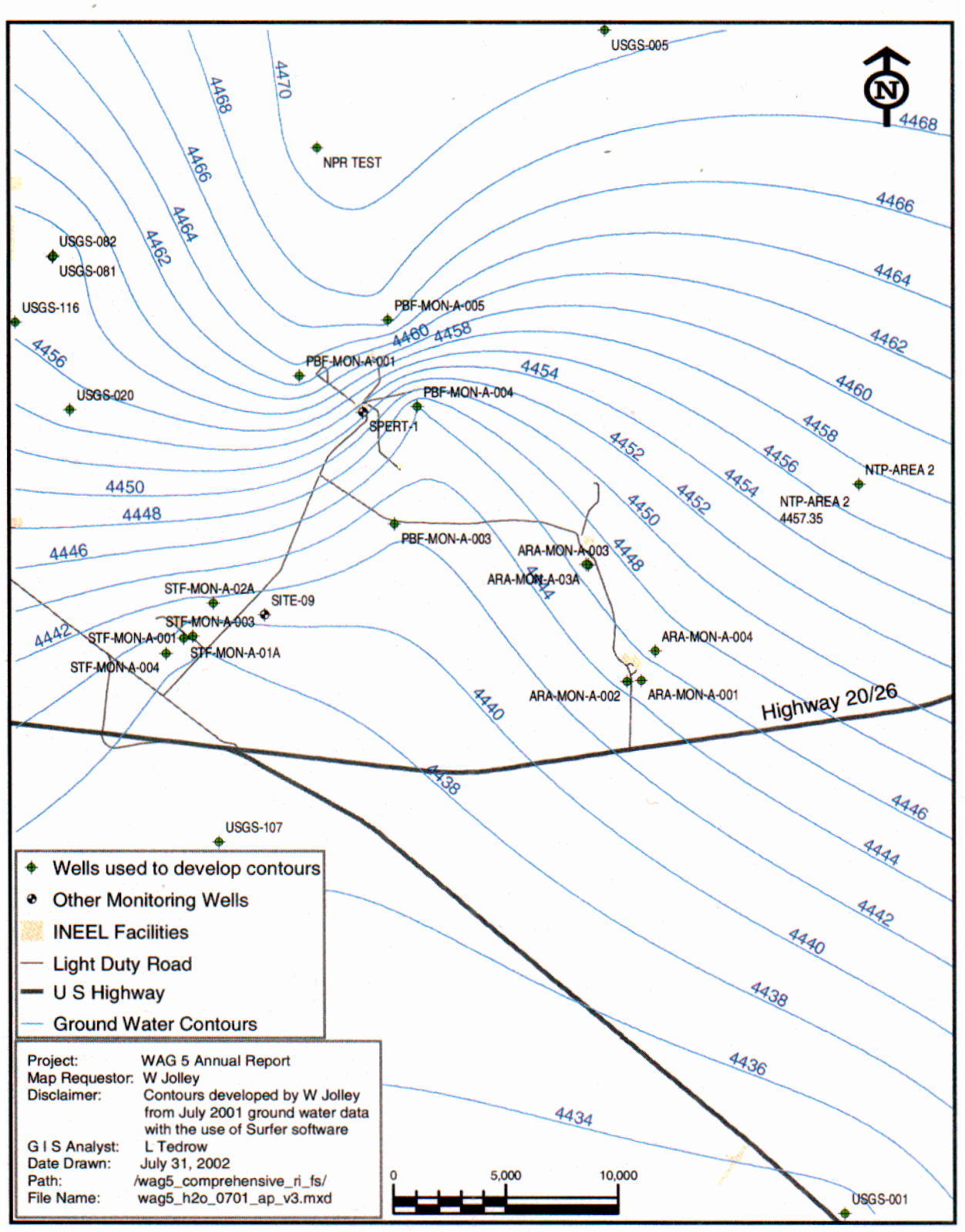

Figure 9. Power Burst Facility/Auxiliary Reactor Area well locations and groundwater elevation contour map.

The need for additional wells, and future monitoring and reporting, would be determined in the process of finalizing a postclosure care and monitoring plan or through submittal of an application for a HWMA/RCRA postclosure permit.

4.1.4.3 Other Institutional Controls. Security of the facility would need to be maintained in accordance with IDAPA 58.01.05.008 (40 CFR 264.14). Entry to the area of the monolith must be controlled, including prevention of entry by livestock that could damage the cover. This standard also requires maintenance of any fences and signs related to access control.

Various postings would be required by HWMA/RCRA and the asbestos disposal requirements from the "National Emissions Standards for Hazardous Air Pollutants" (40 CFR 61). These would include "DANGER KEEP OUT – Hazardous Waste Disposal Site," "Asbestos Waste Disposal Site," and "No Grazing Beyond This Point."

Information regarding the site would be entered in the *Idaho National Engineering and Environmental Laboratory Comprehensive Facility and Land Use Plan* (DOE-ID 1997a) for permanent tracking. In addition, a filing with the Butte County Recorder, regarding the location and description of contaminants, likely would be necessary. Since the property where the monolith is located is described by the land withdrawal documentation, it does not have a deed, so it is not necessary to add deed restrictions reflecting the institutional controls.

The ICP Balance of INEEL Cleanup Project administers long-term stewardship responsibilities for areas where residual contamination remains in place. The administration of potential postclosure care requirements or other institutional controls for PER-620 would be conducted by the Balance of INEEL Cleanup Project. The INEEL implements institutional controls through the *INEEL Sitewide Institutional Controls Plan* (DOE-ID 2004b). This document is updated annually to incorporate new institutional control requirements. These requirements are primarily based on guidance in the "Region 10 Final Policy on the Use of Institutional Controls at Federal Facilities" (EPA 1999). Consistent with the policy, the INEEL is committed to (1) implement measures that ensure short- and long-term effectiveness of institutional controls that protect human health and the environment at federal facility sites undergoing remedial action pursuant to CERCLA (42 USC § 9601 et seq.) and/or corrective action pursuant to RCRA (42 USC § 6901 et seq.), (2) file an initial status report on the status of the institutional controls with DEQ and EPA within 6 months after the signing of any decision documents such as a CERCLA Record of Decision (or Removal Action Memorandum) and/or a RCRA statement of basis, and (3) submit assessment reports at least annually thereafter. The plan describes inspection methods and work control procedures that have been used to institute and inspect the existing INEEL institutionally controlled sites.

Institutional controls (ICs) are used in conjunction with engineered/physical remedies to protect human health and the environment. The "Region 10 Final Policy on the Use of Institutional Controls at Federal Facilities" (EPA 1999) states that institutional controls:

> ... generally include all nonengineered restrictions on activities, access, or exposure to land, groundwater, surface water, waste and waste disposal areas, and other areas or media. Some common examples of tools to implement ICs include restrictions on use or access, zoning, governmental permitting, public advisories, or installation master plans. Institutional controls may be temporary or permanent restrictions or requirements.

4.2 Description of Removal Action Alternatives

4.2.1 Alternative 1—Remove Water in Tanks and Piping; Grout Remaining Substructure and Contents in Place; Remove and Dispose of Abovegrade Structure; Install Cover; and Postclosure Care and Monitoring

Removal and disposal of water in tanks and piping would be performed as described in Section 4.1.1. Grouting the remaining substructure and contents in place would take place as described in Section 4.1.2. Installation of the performance-based cover would take place as described in Section 4.1.2. Removal and disposal of the abovegrade structure would take place as described in Section 4.1.3. Postclosure care and monitoring would take place as described in Section 4.1.4.

4.2.2 Alternative 2—Remove Water in Tanks and Piping; Partially Remove Shielding Lead; Grout Remaining Substructure and Contents in Place; Remove and Dispose of Abovegrade Structure; Install Cover; and Postclosure Care and Monitoring

Removal and disposal of water in tanks and piping would be performed as described in Section 4.1.1. Grouting the remaining substructure and contents in place would take place as described in Section 4.1.2. Installation of the performance-based cover would take place as described in Section 4.1.2. Removal and disposal of the abovegrade structure would take place as described in Section 4.1.3. Postclosure care and monitoring would take place as described in Section 4.1.4.

As compared to Alternative 3, this alternative calls for removal of only that shielding lead and radioactive resin that can be removed without an especially high level of radiation exposure to workers. Further discussion of the criteria used to make this assessment and the evaluation results is contained in Section 5, "Evaluation of Alternatives."

4.2.3 Alternative 3—Remove Water in Tanks and Piping; Remove All Shielding Lead and Activated Components; Grout Remaining Substructure and Contents in Place; Remove and Dispose of Abovegrade Structure; Install Cover

Removal and disposal of water in tanks and piping would be performed as described in Section 4.1.1. Grouting the remaining substructure and contents in place would take place as described in Section 4.1.2. Installation of the performance-based cover would take place as described in Section 4.1.2. Removal and disposal of the abovegrade structure would take place as described in Section 4.1.3.

As compared to Alternative 2, this alternative calls for removal of all lead, activated material, and radioactive resin beds. Further discussion of assessment criteria and evaluation results for this alternative are contained in Section 5, "Evaluation of Alternatives."

A small amount of equipment and facility surface contamination would be left in place. As a result, the substructure and contents would be filled with cementitious grout, and a performance-based cover would be installed.

As compared to Alternatives 1 and 2, the postclosure installation of new monitoring wells, ongoing monitoring, and other related activities is not considered necessary since this alternative would eliminate the nonradionuclide lead inventory and essentially would eliminate the radionuclide material inventory. Institutional controls would be required until the small amount of remaining radionuclides decay to levels that allow for unrestricted access.